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NUREG-0611

Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants

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Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants

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Bulletins and Orders Task Force
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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ABSTRACT

The purpose of this report is to summarize the results of the Bulletins & Orders Task Force generic review of feedwater transients, small break LOCAs, and other Three Mile Island, Unit 2 types of events in Westinghouse-designed operating plants and to confirm the bases for their continued operation. The results of this evaluation are presented in this report in the form of a set of findings and recommendations in each of the principal review areas.

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SUMMARY AND CONCLUSIONS

The accident at Three Mile Island Unit 2 (TMI-2) on March 28, 1979 involved a main feedwater transient coupled with a stuck-open pressurizer power-operated relief valve and a temporary failure of the auxiliary feedwater system. The resulting severity of the ensuing events and the potential generic aspects of the accident on other operating reactors led the NRC to initiate prompt action to: (a) assure that other reactor licensees, particularly those with plants similar in design to TMI-2, took the necessary action to substantially reduce the likelihood for TMI-2-type events and, (b) investigate the potential generic implications of this accident on other operating reactors.

The Bulletins & Orders Task Force was established within the Office of Nuclear Reactor Regulation in early May 1979 and discontinued operations on December 31, 1979. This task force was responsible for reviewing and directing the TMI-2-related staff activities associated with the NRC Office of Inspection and Enforcement bulletins, Commission orders, and generic evaluations of feedwater transients and small break loss-of-coolant accidents for all operating reactors to assure their continued safe operation.

The purpose of this report is to summarize the results of our generic evaluation of feedwater transients, small break loss-of-coolant accidents, and other TMI-2-related events in Westinghouse-designed operating plants and to confirm the bases for their continued operation. The results of this evaluation are presented in this report in the form of a set of findings and recommendations in each of the principal review areas. Additional review of the accident is continuing and further information will be reviewed and modifications will be made, as appropriate.

The generic review of the Westinghouse-designed operating plants has resulted in the following conclusions:

(1) The continued operation of the Westinghouse-designed operating plants is acceptable, provided that certain actions related to the plant's designs and operation, and training of operators identified in this report, are implemented consistent with the recommended implementation schedule.

(2) The actions taken by the licensees with Westinghouse-designed operating plants in response to Office of Inspection and Enforcement Bulletins 79-06A, 79-06A, (Revision 1), and 79-06C (including the actions specified in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurizer Water Reactors") provide added assurance for the protection of the health and safety of the public.

In addition, the Bulletin and Orders Task Force has independently confirmed the safety significance of those related actions recommended by other Office of Nuclear Reactor Regulation task forces discussed in this report.

1. INTRODUCTION

1.1 Background

The accident at Three Mile Island Unit 2 (TMI-2) on March 28, 1979 involved a feedwater transient coupled with a stuck-open pressurizer power-operated relief valve and a temporary failure of the auxiliary feedwater system. The resulting severity of the ensuing events, and the potential generic aspects of the accident on other operating reactors led the NRC to initiate prompt action to: (a) assure that other reactor licensees, particularly those with plants similar in design to TMI-2, took the necessary action to substantially reduce the likelihood for TMI-2-type events, and (b) investigate the potential generic implications of this accident on other operating reactors.

The purpose of this report is to summarize the results of our generic evaluation of feedwater transients, small break loss-of-coolant accidents (LOCAs), and other TMI-2-related events in the Westinghouse (W)-designed operating plants (see Table 1-1) and to confirm the bases for their continued operation. The results of this evaluation are presented in this report in the form of a set of findings and recommendations in each of the principal review areas.

1.2 Bulletins & Orders Task Force

The Bulletins & Orders Task Force (B&OTF) was established within the Office of Nuclear Reactor Regulation (NRR) in early May 1979 and discontinued operations on December 31, 1979. The B&OTF was responsible for reviewing and directing the TMI-2-related staff activities on loss of feedwater transients and small break loss-of-coolant accidents for all operating reactors to assure their continued safe operation. In conducting this activity, the B&OTF concentrated its efforts on the assessment of systems reliability, the review of the analytical predictions of plant performance for both feedwater transients and small break LOCAs, evaluations of generic operating guidelines, the review of emergency plant operating procedures, and the review of operator training.

The B&OTF worked in conjunction with operating plant licensees on plant-specific matters. For the review of generic matters, a working relationship was established with owners groups for plants designed by each nuclear steam supply vendor (Babcock & Wilcox, Westinghouse, Combustion Engineering, and General Electric) and, in some cases, with the individual nuclear steam supply system vendors. At the outset, the highest priority was placed on plants of the Babcock & Wilcox (B&W) design. As short-term actions on these plants were completed, priority was shifted to those pressurized water reactor (PWR) plants manufactured by Westinghouse (W) and Combustion Engineering (CE) and

TABLE 1-1

OPERATING PLANTS WITH WESTINGHOUSE-DESIGNED NUCLEAR STEAM SUPPLY SYSTEMS

	PLANT	UTILITY	POWER LEVEL (thermal megawatts)	DATE OL ISSUED
1.	Haddam Neck	Connecticut Yankee Atomic Power Co.	1825	06/30/67
2.	Yankee Rowe	Yankee Atomic Electric Co.	600	07/09/60
3.	San Onofre 1	Southern California Edison	1347	03/27/67
4.	Prairie Island 1&2	Northern States Power Co.	1650/Unit	08/09/73 & 10/29/74
5.	Farley 1	Alabama Power Co.	2652	06/25/77
6.	Salem 1	Public Service Electric & Gas	3338	08/13/76
7.	North Anna 1	Virginia Electric & Power Co.	2775	11/26/76
8.	D.C. Cook 1&2	Indiana & Michigan Electric Co.	3250, 3391	10/25/74 & 12/23/77
9.	Indian Point 3	Power Authority of the State of New York	2760	12/12/75
10.	Ginna	Rochester Gas & Electric Co.	1520	09/19/73
11.	Kewaunee	Wisconsin Public Service Corp.	1650	12/21/73
12.	Zion 1&2	Commonwealth Edison Co.	3250/Unit	04/06/73 & 11/14/73
13.	Point Beach 1&2	Wisconsin Electric Power Co.	1518/Unit	10/05/70 & 11/10/71
14.	Turkey Point 3&4	Florida Power & Light Co.	2200/Unit	07/19/72 & 04/10/73
15.	Indian Point 2	Consolidated Edison Co.	3025	10/19/71
16.	Trojan	Portland General Electric	3411	11/21/75
17.	H.B. Robinson 2	Carolina Power & Light Co.	2200	07/31/70
18.	Surry 1&2	Virginia Electric & Power Co.	2441/Unit	05/25/72 & 01/29/73
19.	Beaver Valley 1	Duquesne Light Co.	2652	01/30/76

the boiling water reactors (BWRs), a significantly different light water reactor plant type. All BWRs incorporated nuclear steam supply systems (NSSS) designed by the General Electric Company except for one plant, LaCrosse, whose NSSS was designed by Allis Chalmers.

The B&OTF was composed of approximately thirty technical professionals in widely varying disciplines and areas of expertise. The Director of the B&OTF was Dr. D. F. Ross, Jr., and the Deputy Director was T. M. Novak. The members of the B&OTF are listed in Figure 1-1.

The chapter and scope of activities of the B&OTF are discussed in Sections 1.2.1 and 1.2.2 of this report. Section 1.2.3 summarizes the B&OTF activities regarding: (1) the evaluation of responses to the Office of Inspection and Enforcement (IE) bulletins, (2) the issuance and subsequent lifting of Commission Orders issued to licensees with B&W-designed operating reactors, and (3) the evaluation of the system reliability and predicted plant performance for the designs of each of the other reactor vendors with regard to feedwater transients and small break loss-of-coolant accidents. These activities are discussed in greater detail in the appendices to this report.

1.2.1 Charter

The charter of the B&OTF was to review the generic implications of the TMI-2 accident for all operating plants to confirm or establish the bases for their continued safe operation. The end products for this task included:

(1) Safety evaluations and authorizations to resume or continue operations.
(2) Licensing positions regarding the implementation of short-term measures on operating plants.
(3) Recommendations for further improvements in the areas of design and operation, and administrative procedures.
(4) Notification of the Lessons Learned Task Force* of any required actions identified during the B&OTF review.

1.2.2 Scope of Activities

The scope of the B&OTF activities was limited to the review of loss of feedwater transients and small break LOCAs. The specific areas of review related to these events are as follows:

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*The Lessons Learned Task Force was also formed in NRR in response to the TMI-2 accident to identify and evaluate those safety concerns originating with the TMI-2 accident that required licensing actions (beyond those that had been specified in IE Bulletins and Commission Orders) for currently operating reactors, as well as for pending operating license and construction permit applications. For these reasons, the scope of the Lessons Learned Task Force was more general than the scope of the B&OTF.

FIGURE 1-1

BULLETINS & ORDERS TASK FORCE ORGANIZATION

D. F. Ross, Jr., Director
T. M. Novak, Deputy Director

Projects Group*	Systems Group		Analysis Group
W. Kane, Group Leader	S. Israel, Group Leader		Z. Rosztoczy, Group Leader
I. Villalva, Alternate	Section A	Section B	P. Norian, Alternate
P. O'Reilly	G. Mazetis,	P. Matthews,	R. Audette
C. Thomas	Section Leader	Section Leader	B. Sheron
R. Capra	F. Ashe	W. Hodges	W. Jensen
J. Lee, Licensing	W. LeFave	J. Joyce	E. Throm
Assistant	G. Kelly	C. Liang	J. Guttman
	M. Rubin	N. Wagner	R. Frahm
	K. Mahan	T. Greene	
	D. Thatcher	B. Wilson	

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*C. J. Heltemes, Jr., served as group leader of the Projects Group until late September 1979.

(1) Reliability of systems.
(2) Analyses.
(3) Guidelines for the preparation of emergency operating procedures.
(4) Training of operators in emergency operating procedures.

In conducting its activities, the B&OTF had an overlapping responsibility with certain short-term Lessons Learned items as indicated below:

(1) Item 2.1.3.b (Part 1 only) - Instrumentation for Inadequate Core Cooling (Develop Existing Instrumentation)
(2) Item 2.1.7.a - Auto Initiation of Auxiliary Feed
(3) Item 2.1.7.b - Auxiliary Feed Flow Indication
(4) Item 2.1.9 - Transient and Accident Analysis

These items are described in detail in NUREG-0578, "TMI-2 Lessons Learned Task Force Status and Short-Term Recommendations." The B&OTF reviewed the licensees' responses to these short-term Lessons Learned items since these requirements were identified in the generic reviews conducted by the B&OTF.

1.2.3 Summary of Activities

Bulletins

The staff's preliminary review of the TMI-2 accident identified several errors and malfunctions that occurred during the accident and contributed significantly to its severity. As a result, all holders of operating licenses were subsequently instructed to take a number of immediate actions to avoid repetition of these errors. These instructions were specified in a series of bulletins issued by IE.

The initial bulletins defined actions to be taken by licensees of operating plants using B&W-designed nuclear steam supply systems. As the staff's evaluation continued, it was determined that additional actions were necessary, and these bulletins were subsequently expanded, clarified, and issued to all licensees of operating plants for action. For example, holders of operating licenses for B&W-designed reactors were instructed by IE bulletins to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer power-operated relief valve setpoints.

The B&OTF directed the evaluation of each licensee's response to the IE Bulletins. This evaluation process involved an interoffice review group, which included representatives from IE and from the NRR Division of Operating Reactors. When it was concluded that a licensee understood the concerns expressed in the bulletins and provided acceptable responses to the bulletins, the bulletin review was completed and the evaluation issued as a staff report.

The prompt actions taken by licensees in responding to the IE bulletins was considered to be an important contributor to assuring continued safe plant operation. In addition, the bulletins and related evaluations provided substantive input to other staff activities, such as those associated with the generic study efforts and the Lessons Learned Task Force. Thus, many of the subjects addressed by the bulletins were studied in greater depth through other staff activities and studies. Further, the bulletins and the associated responses were used as bases for inspecting plants and auditing reactor operator training.

Orders on Babcock and Wilcox Plants

Soon after the TMI-2 accident, the NRC staff began a reevaluation of the design features of B&W-designed reactors to determine whether additional safety corrections or improvements were necessary. This evaluation involved numerous meetings with B&W and the affected licensees.

The conclusion of these preliminary staff studies was documented in an April 25, 1979 status report to the Commission. It was found that the B&W-designed reactors appeared to be unusually sensitive to certain transient conditions originating in the secondary system. The features of the B&W-designed plants that contributed to this sensitivity were: (1) the relatively small liquid volumes in the secondary side of the steam generators, (2) lack of direct initiation of reactor trip upon the occurrence of off-normal conditions in the feedwater system, (3) reliance on an integrated control system (ICS) to automatically regulate feedwater flow, (4) actuation before reactor trip of a power-operated relief valve on the primary system pressurizer (which, if the valve sticks open, can aggravate the event), and (5) a low steam generator elevation (relative to the reactor vessel) which provides a smaller driving head for natural circulation (except for the Davis-Besse plant).

Because of these features, the B&W designed plants rely more than other PWR designs on the reliability and performance characteristics of the auxiliary feedwater system, the integrated control system, and the emergency core cooling system (ECCS) performance for certain anticipated transients, such as loss of offsite power and loss of normal feedwater. This, in turn, can require greater operator knowledge and skill to safely manage the plant controls during such anticipated transients. As a result of the work supporting the April 25, 1979 report, the NRC staff concluded that certain other short-term design and procedural changes at operating B&W-designed facilities were necessary in order to assure adequate protection to public health and safety.

After a series of discussions between the NRC staff and licensees of operating B&W-designed plants, the licensees agreed to shut down these plants until the actions identified to the Commission in the April 25, 1979 report could be completed. This agreement was confirmed by a Commission Order to each of the licensees. Authorizations to resume operation were issued during the period late May through early July 1979, as individual plants satisfactorily completed the short-term actions and the NRC staff completed an onsite verification of the readiness of the plants to resume operation. In addition to the modifications to be implemented promptly, each licensee also proposed to carry out certain additional long-term modifications to further enhance the capability and reliability of the plant systems to cope with transient events.

Some of the long-term modifications involved the design, procurement, and qualification of safety-grade hardware. Therefore, all of the actions of the long-term portion of the Commission Orders have not yet been completed. NRC staff involvement will continue, to assure that licensees complete each long-term action of the Commission Orders "as promptly as practicable," and that the Commission Orders are closed out by a prompt NRC staff acceptance review.

Generic and Plant-Specific Studies

For B&W-designed operating reactors, an initial NRC staff study was completed and published in NUREG-0560, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company." This study considered the particular design features and operational history of B&W-designed operating plants in light of the TMI-2 accident and related current licensing requirements. As a result of this study, a number of findings and recommendations resulted which are now being pursued.

Generally, the activities involving the B&W-designed reactors are reflected in the actions specified in the Commission Orders. Consequently, as noted earlier, a number of actions have been specified regarding transient and small break LOCA analyses, upgrading of auxiliary feedwater system reliability and performance, procedures for operation action, and operator training. The results of the NRC staff review of the B&W small break analyses are published in NUREG-0565, "Generic Evaluation of Small-Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox-Designed Operating Plants."

Similar studies have been completed for the Combustion Engineering- and General Electric-designed operating plants. These studies, which also focus specifically on the predicted plant performance under different accident scenarios involving feedwater transients and small break loss-of-coolant accidents, are published in NUREG-0635, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants," and NUREG-0626, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications," respectively.

1.3 Conclusions

Based on our generic review of the W-designed operating plants, we have reached the following conclusions:

(1) The continued operation of the W-designed operating plants is acceptable provided that certain actions related to the plants' designs and operation, and training of operators identified in Chapter 3 of this report are implemented consistent with the recommended implementation schedule.

(2) The actions taken by the licensees with W-designed operating plants in response to IE Bulletins 79-06A, 79-06A (Revision 1), and 79-06C (including the actions specified in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors") provide added assurance for the protection of the health and safety of the public.

In addition, the B&OTF independently confirmed the safety significance of those related actions recommended by other NRR task forces discussed in this report.

2. INSPECTION AND ENFORCEMENT BULLETINS (TMI-2 RELATED)

2.1 General

The NRC has a formal program within the Office of Inspection and Enforcement (IE) which informs all licensees of events having safety significance at operating reactors. Whenever an operating plant experiences an event that is of such safety significance as to require action by other licensees, an IE bulletin is issued.

As a result of the Three Mile Island Unit 2 (TMI-2) accident, several IE bulletins were issued. A list of the TMI-2-related bulletins that have been issued to date is provided in Table 2-1.

The actions required of licensees in response to the bulletins listed in Table 2-1 have been classified into two categories: (1) those requiring reviews of the information in the bulletins and subsequent assessments by licensees to determine whether changes in design or procedures are required in light of such information, and (2) those requiring implementation of changes in specific design features or operating procedures at the plants. Each such category is discussed in greater detail in the sections that follow. The status of the NRC staff's evaluation to date of the actions taken by the licensees in responding to the bulletins is provided in Section 2.3.

2.2 Actions Required by IE Bulletins

2.2.1 Review Actions

IE Bulletin 79-05 was the first of a series of bulletins issued in connection with the TMI-2 accident. This bulletin which was issued on April 1, 1979, included a description of the initiating events as well as the course of subsequent events. In addition, this bulletin identified certain actions which had to be taken by licensees having Babcock & Wilcox (B&W)-designed reactors.

IE Bulletin 79-05 served three primary purposes: (1) it informed all nuclear power plant licensees of the events that transpired at TMI-2, (2) it initiated a review by licensees with B&W-designed reactors, and (3) it informed them of the need for certain changes at their plants. Subsequent bulletins [i.e., 79-05A, 79-05C, 79-06, 79-06A, 79-06A (Rev. 1), 79-06C, 79-06B, and 79-08] initiated similar reviews and identified more specific corrective measures to be taken in certain cases. A listing of the general review actions required by the licensees with Westinghouse (W)-designed reactors is presented in this section. These general review actions have been abstracted from IE Bulletins 79-06A, 79-06A (Rev. 1), and 79-06C, the bulletins applicable to W-designed reactors. Actions required by the bulletins that involve specific changes to the plant design or operating procedures are discussed in Section 2.2.2.

TABLE 2-1
LISTING OF TMI-2-RELATED IE BULLETINS

Bulletin	Subject	Issue Date	Issued to Licensees
79-05	Nuclear Accident at Three Mile Island	4/1/79	All B&W-designed power reactors with an operating license for action and all other power reactors for information.
79-05A	Nuclear Accident at Three Mile Island - Supplement	4/5/79	All B&W-designed power reactors with an operating license for action and all other power reactors for information
79-06	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Accident	4/11/79	All pressurized water reactors with an operating license (except B&W-designed) for action and all other power reactors for information
79-06A	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Accident (replaced 79-06)	4/14/79	All Westinghouse-designed power reactors with an operating license for action and all other power reactors for information
79-06B	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Accident	4/14/79	All Combustion Engineering (CE)-designed power reactors with an operating license for action and all other power reactors for information
79-08	Events Relevant to Boiling Water Power Reactors Identified During Three Mile Island Accident	4/14/79	All boiling water reactors with an operating license for action and all other power reactors for information
79-06A (Rev. 1)	Review of Operational Error and System Misalignment Identified During the Three Mile Accident. (Revised Items 3 and 13 of Bulletin 79-06A)	4/18/79	All Westinghouse-designed power reactors with an operating license for action and all other power reactors for information
79-05B	Nuclear Accident at Three Mile Island	4/21/79	All B&W-designed reactors with an operating license for action and all other power reactors for information
79-05C & 79-06C	Nuclear Accident at Three Mile Island - Supplement	7/26/79	All B&W, W, and CE-designed power reactors with an operating license for action and all other reactors for information

The items that follow were incorporated in IE Bulletin 79-06A by reference to IE Bulletins 79-05 and 79-05A:

(1) Review the description of circumstances surrounding the TMI-2 accident. This review should be directed toward understanding (a) the seriousness and consequences of simultaneously blocking both auxiliary feedwater trains, (b) the apparent operational errors which led to the eventual core damage, (c) the potential which exists for having a water level in the pressurizer simultaneously with the reactor not being full of water, and (d) the necessity to systematically analyze plant conditions and parameters in order to take appropriate corrective action.

(2) Review operating procedures for coping with transients and accidents to assure that they acknowledge the possibility of forming voids in the primary coolant system large enough to compromise core cooling capability, especially natural circulation capability, and that they identify (a) operator actions required to prevent formation of such voids, and (b) operator actions required to enhance core cooling in the event such voids are formed.

(3) Review operating procedures and training instructions to assure that operators do not override automatic actions of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions.

(4) Review all safety-related valve positions and procedures for positioning valves, including those for use following maintenance, testing, plant and system startup and supervisory periodic surveillance to assure that they will be in the correct position during all operational modes.

(5) Review the operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the containment to assure that the transfer will not occur inadvertently.

(6) Review operating modes and procedures to deal with significant amounts of hydrogen gas generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

2.2.2 Changes to Plant Design Features and Operating Procedures

In the days immediately following the issuance of IE Bulletin 79-05, the NRC received additional preliminary information related to the TMI-2 incident. Based on this information, the NRC identified six deficiencies consisting of human errors, design inadequacies, and mechanical failures as the likely candidates that led to the core damage and radiation releases at Three Mile Island. To assure that all licensees were fully informed of these factors, followup bulletins were issued, beginning with IE Bulletin 79-05A on April 5, 1979 for licensees in B&W-designed reactors and terminating with IE Bulletins 79-05C and 79-06C for licensees with B&W, W- and CE-designed reactors, respectively, as indicated in Table 2-1.

In contrast to IE Bulletin 79-05, these later bulletins not only provided information for licensees to review, but also identified specific action to be taken to reduce the likelihood of the type of event which occurred at TMI-2. The following is a listing of the types of actions to be taken by licensees with W-designed plants:

(1) The licensees were required to revise existing operating procedures to specify that, if the high pressure injection (HPI) system has been automatically actuated because of a low reactor coolant system pressure condition, it must remain in operation until either:

    (a) Both low pressure injection system pumps are operating and flow has been observed for 20 minutes or longer at a rate which would assure stable plant behavior, or

    (b) The high-pressure injection (HPI) system has been operating for 20 minutes and all hot and cold leg temperatures are at least 50°F below the saturation temperature for the existing reactor coolant system pressure. If 50°F subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50°F and the length of time HPI is in operation shall be limited by the pressure/temperature considerations for the vessel integrity.

(2) The licensees were required to modify existing operating procedures to assure that the operators would not rely solely upon pressurizer level indication alone, but that they would consider other plant parameters in evaluating plant conditions, such as water inventory in the reactor primary system.

(3) The licensees were required to modify the existing containment isolation system design and operating procedures, as necessary, to permit the isolation upon automatic initiation of safety injection of those lines whose isolation would not degrade safety features or cooling capability.

(4) The licensees were required to modify maintenance and test procedures, as necessary, to assure the operability of redundant safety-related systems prior to their removal from service and following maintenance or testing. Explicit notification is to be given to all reactor operational personnel whenever a safety-related system is removed from and returned to service.

(5) The licensees were required to modify reporting procedures for prompt NRC notification to assure that the NRC is notified within one hour of the time that a reactor is not in a controlled or expected condition of operation. Further, at that time, an open continuous communication channel with the NRC was required to be established and maintained.

(6) Licensees with plants with pressurizer power-operated relief valves (PORVs) were required to prepare and implement immediately specific procedures which identify those plant indications that the operators may utilize to determine that the PORV(s) are open and direct the operators to take action to secure the PORV by closing the PORV block valve when the reactor coolant system pressure is reduced below the set point for normal automatic closure of the PORV and the PORV remains stuck in the open position.

(7) Licensees with plants that use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system were required to trip the low pressurizer level setpoint bistables such that, when the pressurizer pressure reaches the low setpoint, safety injection would be initiated regardless of the pressurizer level. For these particular plants, the pressurizer level bistables may be returned to their normal operating positions during the pressurizer pressure channel surveillance tests. In addition, operators were instructed to manually initiate safety injection when the pressurizer pressure indication reaches the actuation setpoint, whether or not the level indication has dropped to the actuation setpoint.

(8) Licensees with plants where the auxiliary feedwater system is not automatically initiated were instructed to prepare and implement immediately procedures requiring the stationing of an individual whose function is to promptly initiate adequate auxiliary feedwater flow to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action. This individual should have no other assigned concurrent duties and should be in direct and continuous communication with the control room.

(9) The licensees were required to propose changes to those technical specifications which must be modified as a result of implementing the above items and to identify design changes necessary to effect long-term resolution of these items.

(10) In IE Bulletins 79-05C and 79-06C [which superseded item 4(c) of Bulletin 79-05A, item 7(c) of Bulletin 79-06A, and item 6(c) of Bulletin 79-06B], licensees were required to take the following immediate actions:

(a) Upon reactor trip and initiation of HPI caused by low reactor coolant system pressure, all operating reactor coolant pumps shall be tripped immediately.

(b) Two licensed operators shall be provided in the control room at all times during operation to accomplish action (a) above, as well as other immediate and necessary followup actions.

(11) IE Bulletins 79-05C and 79-06C also required the following short-term actions:

(a) The licensees were required to perform and submit a report of loss-of-coolant (LOCA) analyses for their plants covering a range of small break sizes and a range of time lapses between reactor trip and pump trip. The resulting peak cladding temperature was to be determined for each pair of values of the parameters. The range of values for each parameter must be wide enough to assure that the maximum peak cladding temperature or, if appropriate, the region containing peak cladding temperatures greater than 2200°F is identified.

(b) The licensees were required to develop new guidelines, for operational procedures based on the analyses done under item (a) above, for both LOCA and non-LOCA transients that take into account the impact of reactor coolant pump trip requirements. Specifically, for B&W-designed reactors, these guidelines should include appropriate requirements to fill the steam generators to a higher level, following reactor coolant pump trip, to promote natural circulation flow.

(c) The licensees were required to revise emergency procedures and train all licensed reactor operators and senior reactor operators based on the guidelines developed under item (b) above.

(d) The licensees were required to provide analyses and develop guidelines and procedures related to inadequate core cooling (as discussed in Section 2.1.9 of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations") and define the conditions under which a restart of the reactor coolant pumps should be attempted.

(12) As a long-term action, the licensees were required to propose and submit a design that will assure automatic tripping of the operating reactor coolant pumps under all circumstances in which this action may be needed.

2.3 Evaluation of Licensee Responses to IE Bulletins

Our evaluation of all responses to the TMI-2-related IE Bulletins from licensees with W-designed operating plants has been completed. Requests for further information and clarification of individual licensee's responses were made to facilitate the review of these responses. Meetings were held with the individual licensees and their representatives, as necessary, to expedite the review effort. In addition, licensees with W-designed operating plants formed an owners group to interact with the staff to resolve several issues raised by the IE Bulletins. Evaluation reports are being prepared for transmittal to each licensee reporting the results of the review effort. These reports state our conclusions about each licensee's responsiveness to the staff's concerns. Each report also indicates whether the licensee: (1) has correctly interpreted the IE Bulletin, (2) has demonstrated its understanding of the salient issues arising from the TMI-2 incident and its implications for the licensee's own operations, and (3) has provided the necessary added assurance for the protection of the public health and safety during plant operation.

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3. GENERAL FINDINGS AND RECOMMENDATIONS

This section contains statements of concern (or findings) and recommendations. The purpose of this section is to identify our recommendations along with the specific appendices where further detail is included. A schedule for implementation of the recommendations may be found in Table 3-1.

Based on the results of our review of feedwater transients and other related events, we have concluded that the Westinghouse (W)-designed operating plants can continue to operate pending incorporation of the recommended changes identified in this report. The bases for this conclusion are as follows:

(1) Westinghouse-designed plants are much less sensitive to feedwater transients than Babcock and Wilcox-designed plants were in their pre-TMI-2 configuration.

(2) Changes in plant procedures and operator training and design modifications have been required by IE Bulletins 79-06A, 79-06A, Revision 1, and 79-06C* based on the experience at TMI-2 which:

   (a) Eliminate the reactor trip coincidence logic on low pressure-low pressurizer level.
   (b) Provide added assurance that engineered safety features operation will not be terminated prematurely.
   (c) Reduce the susceptibility of manually actuated auxiliary feedwater systems to operator error or inaction by requiring a dedicated operator to activate the auxiliary feedwater system.
   (d) Increase operator awareness and understanding of the events that occurred at TMI-2 through retraining programs.
   (e) Assure that more frequent checks are made to verify that valves and safety-related systems are maintained in correct position and also that they are returned to their correct positions following test and/or maintenance.
   (f) Provide additional assurance that the occurrence of a stuck-open power-operated relief valve will be detected and isolated in a timely manner.

   (g) Require a dedicated operator to trip the reactor coolant pumps in case of a reactor trip and initiation of high pressure injection caused by low reactor coolant system pressure (until the automatic pump trip required by NUREG-0623 has been installed according to the implementation schedule specified in that report).

(3) The NRC's Lessons Learned Task Force has developed a more comprehensive set of short-term actions (specified in NUREG-0578) which the NRC staff has implemented as requirements for operating plants.

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*As amended by the actions specified in NUREG-0623.

(4) The Lessons Learned Task Force has also developed a number of long-term recommendations which have been published in NUREG-0585 regarding changes in nuclear plant and operation and the regulatory process.

3.1 Auxiliary Feedwater Systems

3.1.1 Summary

The Three Mile Island Unit 2 (TMI-2) accident and subsequent investigations and studies highlighted the importance of the auxiliary feedwater (AFW) system in the mitigation of transients and accidents. As part of our assessment of the TMI-2 accident and related implications for operating plants, we evaluated the AFW systems for all operating plants having nuclear steam supply systems (NSSS) designed by W (25 units) and CE (8 units). (See Note below.)

The objectives of this study were to: (1) identify necessary changes in AFW system design or related procedures at these plants, in order to assure their continued safe operation, and (2) to identify other system characteristics in the AFW system design of these plants which, on a long-term basis, may require system modifications. To accomplish these objectives, we:

(1) Reviewed plant-specific AFW system designs in light of current regulatory requirements,

(2) Assessed the relative reliability of the various AFW systems under various loss of feedwater transients, one of which was the initiating event at TMI-2, and other postulated potential failure conditions by determining the potential for AFW system failure due to common causes, single point vulnerabilities and human error.

As part of our evaluation, we performed a standard deterministic type of safety review, using as principal guidance the acceptance criteria specified in Section 10.4.9 of the Standard Review Plan (SRP). In conjunction with this deterministic review, we used event tree and fault tree logic techniques, as part of a reliability analysis to determine dominant failure modes and assess AFW system comparative reliability levels under specified types of transients. When the recommendations identified in this review are implemented, the reliability of the AFW systems for each operating plant should be improved to a degree dependent on whether the comparative reliability could be characterized as being relatively high or relatively low.

This section presents in summary form the results of the generic AFW system review and recommendations that should be implemented to improve the performance and reliability

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Note: Studies of the AFW systems at Babcock and Wilcox (B&W)-designed operating plants were the subjects of separate Commission Orders and other work performed by the staff.

of the AFW systems of the various W-designed operating plants. Details of our review are contained in Appendix III. Appendix X provides a separate AFW system description, evaluation, and recommendations for each individual plant.

3.1.2 Recommendation Categories

The recommendations resulting from this review are categorized as generic and plant-specific, as well as short-term and long-term. The generic recommendations (designations GS and GL refer to short- and long-term, respectively) are a result of similarities in AFW system potential problems between plants, and are applicable to more than one plant. The generic recommendations and the concerns which led to these recommendations are described in this section. There are also plant-specific recommendations that are unique to a given plant's AFW system. The plant-specific recommendations are addressed more fully in the individual plant evaluations in Appendix X. The individual plant evaluations have already been transmitted to each licensee as required modifications to the AFW system design or associated procedures. This was done so that implementation of the required actions could proceed in a timely manner.

The short-term recommendations represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable. This implementation schedule is intended to be consistent with the schedule for implementation of the requirements specified in NUREG-0578. If conflicts should arise, the schedule specified in NUREG-0578 is more restrictive and thus takes precedence.

There are two significant limitations of the AFW system review and evaluation which should be noted, as well as their effect on the recommendations.

(1) While our review covered the classification and divisional redundancy of power sources for AFW system equipment and instrumentation and controls, and the type of instrumentation and controls provided for the overall AFW system, we did not attempt to review detailed logic and control diagrams. This explains in part the conservative approach we used in applying to all plants the short- and long-term generic recommendations GS-7 and GL-5, which deal with non-redundant and non-Class 1E circuitry for AFW system automatic initiation systems.

(2) The review is not considered to be a complete evaluation of postulated high energy pipe breaks that could affect the AFW system, since piping isometric and plant arrangement drawings were not reviewed. However, where system flow sheets revealed potential pipe breaks that could cause total loss of AFW system capability, these problem areas have been identified and included in the long-term recommendations for further evaluation.

3.1.3 Short-Term Generic Recommendations

3.1.3.1 Technical Specification Time Limit on AFW System Train Outage

Concern

Several of the plants reviewed have plant Technical Specifications that permit one of the AFW system trains to be out of service for an indefinite time period. Indefinite outage of one train reduces the defense-in-depth provided by multiple AFW system trains.

Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

3.1.3.2 Technical Specification Administrative Controls on Manual Valves - Lock and Verify Position

Concern - Several of the plants reviewed use a single manual valve or multiple valves in series in the common suction piping between the primary water source and the AFW system pump suction. At some plants the valves are locked open, while at others, they are not locked in position. If the valves are inadvertently left closed, the AFW system would be inoperable because the water supply to the pumps would be isolated. Since there is no remote valve position indication for these valves, the operator has no immediate means of determining valve position.

Further, the Technical Specifications for plants with locked-open manual valves do not require periodic inspection to verify that the valves are locked and in the correct position. For most plants where the valves are not locked open, valve position is verified on some periodic basis.

Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

3.1.3.3 AFW System Flow Throttling-Water Hammer

Concern - Several of the plants reviewed apparently throttle down the AFW system initial flow to eliminate or reduce the potential for water hammer. In such cases, the overall reliability of the AFW system can be adversely affected.

Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

3.1.3.4 Emergency Procedures for Initiating Backup Water Supplies

Concern - Most of the plants do not have written procedures for transferring to alternate sources of AFW supply if the primary supply is unavailable or exhausted. Without specific criteria and procedures for an operator to follow to transfer to alternate water sources, the primary supply could be exhausted and result in pump damage or a long interruption of AFW flow.

Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

(1) The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated, and

(2) The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3.1.3.5 Emergency Procedures for Initiating AFW Flow Following a Complete Loss of Alternating Current Power

Concern - Some operating plants depend on ac power for all sources of AFW system supply, including the turbine-driven pump train. In the event of loss of offsite and onsite ac power, ac power-dependent lube oil supply or lube oil cooling for the pump will stop, and/or manual actions are required to initiate AFW flow from the turbine-driven pump by manually opening the turbine steam admission valve and/or AFW system flow control valves. There are no procedures available to the plant operators for AFW system initiation and control under these conditions. This could result in a considerable time delay for AFW system initiation, since the operators would not be guided by procedures dealing with this event.

Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any ac power source. If manual AFW system initiation or flow control is required following a complete loss of ac power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on ac power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all ac power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until ac power is restored. Adequate lighting powered by direct current (dc) power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

3.1.3.6 AFW System Flow Path Verification

Concern - Periodic testing of the AFW system is accomplished by testing of individual components of one flow train (periodic pump recirculation flow test or automatic valve actuation), thus altering the normal AFW system flow path(s). The flow capability of the entire AFW system, or at least one integral AFW system train, is only demonstrated on system demand following a transient, or if the AFW system is used for normal plant startup or shutdown.

Recent Licensee Event Reports indicate a need to improve the quality of system testing and maintenance. Specifically, periodic testing and maintenance procedures inadvertently result in (1) more than one AFW system flow train being unavailable during the test, or (2) the AFW system flow train under test not being properly restored to its operable condition following the test or maintenance work. The Office of Inspection and Enforcement has taken action to correct Item (1); the recommendation below is made to correct Item (2).

Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

(1) Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

(2) The licensee should propose Technical Specifications to assure that, prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

3.1.3.7 Non-Safety Grade, Non-Redundant AFW System Automatic Initiation Signals

Concern - Some plants with an automatically initiated AFW system utilize some initiation signals that are not safety-grade, do not meet the single failure criterion, and are not required by the Technical Specifications to be tested periodically. This can result in reduced reliability of the AFW system.

Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer-term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

(1) The design should provide for the automatic initiation of the auxiliary feedwater system flow.

(2) The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

(3) Testability of the initiation signals and circuits shall be a feature of the design.

(4) The initiation signals and circuits should be powered from the emergency buses.

(5) Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

(6) The ac motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

(7) The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

3.1.3.8 Automatic Initiation of AFW Systems

Concern - For plants with a manually initiated AFW system, there is the potential for failure of the operator to manually actuate the system following a transient in time to maintain the steam generator water level high enough to assure reactor decay heat removal via the steam generator(s). While IE Bulletin 79-06A requires a dedicated individual for W-designed operating plants with a manually initiated AFW system, further action should be taken in the short-term. This concern is identical to Item 2.1.7.a of NUREG-0578.

Recommendation GS-8 - The licensee should install a system to automatically initiate AFW system flow. This system need not be safety-grade; however, in the short-term, it should meet the criteria listed below, which are similar to Item 2.1.7.a of NUREG-0578. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements, as indicated in Recommendation GL-2.

(1) The design should provide for the automatic initiation of the auxiliary feedwater system flow.

(2) The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

(3) Testability of the initiating signals and circuits should be a feature of the design.

(4) The initiating signals and circuits should be powered from the emergency buses.

(5) Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

(6) The ac motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

(7) The automatic initiation signals and circuits should be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

3.1.4 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins & Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system design in W- and CE- designed operating plants. They have not been examined for specific applicability to individual W- and CE-designed operating plants.

3.1.4.1 Primary AFW Water Source Low Level Alarm

Concern - Plants which do not have level indication and alarm for the primary water source may not provide the operator with sufficient information to properly operate the AFW system.

Recommendation - The licensee should provide redundant level indication and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

3.1.4.2 AFW Pump Endurance Test

Concern - Since it may be necessary to rely on the AFW system to remove decay heat for extended periods of time, it should be demonstrated that the AFW pumps have the capability for continuous operation over an extended time period without failure.

*Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3.1.4.3 Indication of AFW Flow to the Steam Generators

Concern - Indication of AFW flow to the steam generators is considered important to the manual regulation of AFW flow to maintain the required steam generator water level. This concern is identical to Item 2.1.7.b of NUREG-0578.

**Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

(1) Safety-grade indication of AFW flow to each steam generator should be provided in the control room.

(2) The AFW flow instrument channels should be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for

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*Based on the results of recent tests, the 72-hour specification has been reduced to 48 hours.

**The implementation of this requirement was modified by H. R. Denton's October 30, 1979 letter to all operating reactor licensees.

the AFW system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

3.1.4.4 AFW System Availability During Periodic Surveillance Testing

Concern - Some plants require local manual realignment of valves to conduct periodic pump surveillance tests on one AFW system train. When such plants are in this test mode and there is only one remaining AFW system train available to respond to a demand for initiation of AFW system operation, the AFW system redundancy and ability to withstand a single failure are lost.

Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train, and which have only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system from the test mode to the operational alignment.

3.1.5 Long-Term Generic Recommendations

3.1.5.1 Automatic Initiation of AFW Systems

Concern - This concern is the same as short-term generic recommendation GS-8; namely, failure of an operator to actuate a manual start AFW system in time to maintain steam generator water level high enough to assure reactor decay heat removal via the steam generator(s).

Recommendation GL-1 - For plants with a manual starting AFW system, the licensee should install a system to automatically initiate the AFW system flow. This system and associated automatic initiation signals should be designed and installed to meet safety-grade requirements. Manual AFW system start and control capability should be retained with manual start serving as backup to automatic AFW system initiation.

3.1.5.2 Single Valves in the AFW System Flow Path

Concern - This concern is the same as short-term generic recommendation GS-2; namely, AFW system inoperability due to an inadvertently closed manual valve that could interrupt all AFW system flow.

Recommendation GL-2 - Licensees with plant designs in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path, should install redundant parallel flow paths (piping and valves).

Licensees with plant designs in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions into the surveillance requirements.

3.1.5.3 Elimination of AFW System Dependency on Alternating Current Power Following A Complete Loss of Alternating Current Power

Concern - This concern is the same as short-term generic recommendation GS-5 - namely, delay in initiation of AFW system operation or maintaining AFW system operation following a postulated loss of onsite and offsite ac power; i.e., ac power blackout.

Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any ac power source for at least two hours. Conversion of dc power to ac power is acceptable.

3.1.5.4 Prevention of Multiple Pump Damage Due to Loss of Suction Resulting From Natural Phenomena

Concern - In many of the operating plants, the normal water supply to the AFW system pumps (including the interconnected piping) is not protected from earthquakes or tornadoes. Any natural phenomenon severe enough to result in a loss of the water supply could also be severe enough to cause a loss of offsite power with loss of main feedwater, resulting in an automatic initiation signal to start the AFW system pumps. The pumps would start without any suction head, leading to cavitation and multiple pump damage in a short period of time, possibly too short for the operators to take action that would protect the pumps. This may lead to unacceptable consequences for some plants, due to a complete loss of feedwater (main and auxiliary).

Recommendation GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure, or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

3.1.5.5 Non-Safety Grade, Non-Redundant AFW System Automatic Initiation Signals

Concern - This concern is the same as short-term generic recommendation GS-7 - namely, reduced AFW system reliability as a result of use of non-safety-grade, non-redundant signals, which are not periodically tested, to automatically initiate the AFW system.

Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

3.1.6 Plant Specific AFW System Recommendations

The short-term and long-term plant-specific recommendations applicable to the AFW systems for each plant are identified and discussed in Appendix X.

3.2 Analysis

Slow system depressurization accidents resulting from small breaks in the primary system have not, until recently, been subjected to detailed analytical study comparable to that devoted to large breaks. Typically, small breaks have been analyzed down to the smallest break size that would produce system depressurization without uncovering the core in accordance with the single failure criterion and other requirements imposed by Appendix K to 10 CFR Part 50. These analyses assumed the availability of heat removal through the steam generators following reactor scram, power loss to the reactor coolant pumps upon scram, and normal plant protective and emergency core cooling systems activation initiated by the system depressurization. While the analyses, in general, were sufficient to show compliance with the requirements of Section 50.46 of 10 CFR Part 50 (10 CFR § 50.46), they failed to provide the necessary information needed for operator action following a small break, as was pointed out by the NRC's Lessons Learned Task Force in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

Reassessment of the failure modes assumed in small break accidents as a result of recent events, particularly in light of the TMI-2 accident, have led the staff and industry to a considerably broader interpretation of potential accident scenarios than held previously.

Basically, our review focused on the information presented in the Westinghouse report, WCAP-9600, "Report on Small Break Accident for Westinghouse NSSS System," as supplemented. This report was submitted for our review by the Westinghouse Operating Plants Owners Group. Our review included the following considerations: analytical methods, pressurizer model, steam generator model, non-condensible gases, experimental verification, break discharge model, vessel mixture level, and system nodal detail. The results of our review of this report are summarized below. Further discussion of the findings on each of the principal areas of concern and recommendations for further improvements (other than those identified in this section) may be found in Appendix VIII.

3.2.1 Small Break LOCA Analyses

Finding

The small break LOCA analysis methods used by W are satisfactory for the purpose of predicting trends in plant behavior following a small break LOCA. The results of the analyses can be used to develop improved emergency procedures, and to train reactor operators. However, several individual analytical models identified in Section 4.2.1 of Appendix VIII require improvement or further confirmation. In addition, comparison of the total analysis method with available small break integral test data (Semiscale Test S-02-6) has indicated large uncertainties in the calculations. The analysis methods should be revised and verified before they can be considered for NRC approval under 10 CFR § 50.46.

Recommendations

(a) The analysis methods used by Westinghouse for small break LOCA analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including, data from the LOFT and Semiscale facilities.

(b) Plant-specific calculations using the NRC-approved model for small break LOCAs as described in (a) above, to show compliance with 10 CFR § 50.46 should be submitted for NRC approval by all licensees.

3.2.2 Role of Non-Safety Equipment in Mitigating Small Break LOCAs

Finding

Westinghouse has performed a sufficient spectrum of small break LOCA analyses to identify the anticipated system performance for breaks in this range. These analyses provide adequate bases for developing improved operator guidelines, and demonstrate that operator action and a combination of heat removal by the steam generators, high pressure injection system, and the break ensure adequate core cooling. The required operator actions are the following: tripping the reactor coolant pumps shortly after occurrence of a LOCA, MPI termination in the event of a repressurization, and manual restoration of AFW or PORV opening in case of a loss of all feedwater. Pump trip is required because W calculations show that, for a narrow range of small break sizes, the 10 CFR § 50.46 limits on peak cladding temperature could be exceeded if the pumps are not tripped. According to W estimates, at least 10 minutes are available for the operator to perform this action. Our evaluation of the W analysis (published in NUREG-0623) indicates that the times available for the operator could be shorter than 10 minutes, and may even be as short as three minutes, indicating a need for automatic actuation.

System repressurization can occur for the smallest breaks following HPI actuation. When appropriate plant conditions are reached, as specified in the guidelines for emergency procedures, operator termination of HPI is permissible to avoid filling the pressurizer with water and challenging the PORV.

If, in addition to the small break LOCA, feedwater flow (both main feedwater and auxiliary feedwater) is lost, or if, for any reason natural circulation is not established, there will eventually (after at least 25 minutes, as discussed in Appendix VII) be no heat removal through the steam generators. In this case, operator action is required to restore feedwater flow, or to open the pressurizer relief valves and block valves (if closed). According to W, in case of a complete loss of feedwater flow, either action will serve to depressurize the primary system, so that sufficient safety injection flow can be established. If natural circulation fails, the operator must open the pressurizer relief valves. Westinghouse indicated that approximately one hour is available for the operators to reinitiate feedwater flow. Opening of the relief valves must be accomplished within 40 minutes in order to keep the consequences of the event within acceptable limits. Our review of the W calculations revealed that W overestimated the relief valve flow rate used in the calculations, and that the PORVs should be opened sooner. The importance of the flow rate through relief and safety valves is recognized in NUREG-0578, Section 2.1.2, which requires that full-scale prototype tests be performed by July 1981. Based on the previous discussion, we have concluded that a diverse decay heat removal path, independent of the steam generators, is desirable.

Recommendations

(a) Tripping of the reactor coolant pumps in case of a LOCA is not an ideal solution. The licensees should consider other solutions to the small break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small break LOCA. The signals designated to initiate the pump trip should be carefully selected in order to differentiate between a small break LOCA and other events which do not require reactor coolant pump trip as discussed in NUREG-0623.

(b) The Westinghouse small break LOCA analyses relied on equipment which has not previously been characterized as part of the reactor protection system or part of the engineered safety features. The equipment used to provide reactor coolant pump trip, the pressurizer spray valves, the pressurizer relief valves, the pressurizer relief block valves, the equipment used to automatically actuate the pressurizer relief valves, and the equipment used to remotely control the pressurizer relief and block valves fall into this category. The reliability and redundancy of these systems should be reviewed and upgraded, if needed, to comply with the requirements of Section 9 of NUREG-0585 regarding the interaction of non-safety and safety-grade systems.

3.2.3 Michelson's Concerns

Finding

A number of concerns related to decay heat removal following a very small break LOCA, and other related items, were identified by Mr. C. Michelson of TVA (see Section 4.1 of Appendix VIII). These concerns were identified for PWRs designed by Babcock & Wilcox and Combustion Engineering. Westinghouse has reviewed these concerns and provided an analysis in WCAP-9600 of those items that relate to plants of their design. Postulated modes of two-phase flow natural circulation play an important role in the W analysis. The analysis provides an adequate assessment of these concerns. However, experimental results are not available to support the analytical predictions.

Recommendations

(a) The various modes of two-phase flow natural circulation which are expected to play a significant role in plant response following a small break LOCA should be demonstrated experimentally. The results of the tests should be submitted for NRC review.

(b) Appropriate means, including additional instrumentation if necessary, should be provided in the control room to facilitate verifying whether natural circulation has been established.

3.2.4 PORV Failures in W-Designed Plants

The incident at TMI-2 was initiated by a loss of feedwater transient, which caused a pressure increase in the primary system. This caused the subsequent opening of a PORV at the top of the pressurizer. The failure of the PORV to reseat when the primary system was depressurized below the valve setpoint pressure, together with the failure of the operator to isolate the PORV, produced the equivalent of a small break LOCA. The expected frequency of a stuck-open PORV has been evaluated by two methods in this study: a) using operating experience, and b) using analytical predictions of plant transient responses. The former evaluation is based on a summary of operational data provided by W and our evaluation of W-designed plant operating history contained in NUREG-0618, while the latter evaluation is based on analyses of anticipated transients provided in WCAP-9600.

Findings

(a) The record of PORV failures (to close) for all PWRs, 13 in approximately 200 reactor years, has demonstrated a potential cause of a small break LOCA. Ten PORV failures have occurred at B&W-designed plants, whereas two have been recorded for Westinghouse-designed plants, and one at a Combustion Engineering-designed plant. We have no evidence that this record is complete, nor do we have evidence to the contrary. In attempting to estimate the probability of small break LOCAs produced by PORV failure, we have evaluated data from the history of PORV openings at Westinghouse-designed plants and the frequency of overpressure transients at these plants. We have also evaluated analytical predictions of PORV openings for these transients provided by W. We cannot conclude that the set of PORV opening data is sufficiently complete to permit accurate quantification of the probability of the valve failing to close.

Currently available results of analytical calculations with the LOFTRAN code cannot be used to quantify valve actuation probabilities because of the effect of certain conservative input, as discussed in Section 3.3 of Appendix VIII. (See Recommendation (3.2.4.e) below).

(b) Analyses performed by W have demonstrated that the anticipatory reactor trip on turbine trip will prevent the PORV from opening. Preliminary information (Table VIII-2) indicates that it is available in all W-designed operating plants.

Recommendation

(a) Licensees with W-designed operating plants should confirm that their plants have an anticipatory reactor trip on turbine trip. We recommend that the licensee of any plant where this trip is not present provide a conceptual design and evaluation for the installation of this trip.

Finding

The Pressure Integral Derivative (PID) controller, installed on at least one PORV in most W-designed plants, has caused spurious valve actuations. Westinghouse has recommended that the PID controller be modified in plants where it is used for PORV actuation by raising the interlock to the same value as the PID control bistable, in order to eliminate spurious openings.

Recommendation

(b) The above-described modification should be implemented by affected licensees.

Finding

Some W-designed operating plants want to modify the anticipatory trip (reactor trip upon turbine trip) so that it operates at power levels of 50% and above, instead of the present operational level of 10% and above. Since many PORV openings have occurred at low power, this modification may increase the probability of a small break LOCA resulting from the failure of an opened PORV to close.

Recommendation

(c) This proposed anticipatory trip modification should not be made until it has been shown on a plant-by-plant basis that the small break LOCA probability resulting from a stuck-open PORV is little affected by the modification.

Finding

A PORV supplied by Control Components, Inc. (CCI) used in the McGuire plant (owned by Duke Power) failed during hot functional testing. Because this valve is different from the Copes-Vulcan design, which comprises the operational data for W-designed plants, its failure mechanism and failure rate must be determined to be equitable with thet of the Copes-Vulcan valves, in order to include both in the same population. At present, a data base for operational failures for this valve does not exist.

Recommendation

(d) Any plant using or planning to use this valve without modification should provide complete justification for such use in light of this failure. This matter should be addressed on a plant-by-plant basis. The valve should be modified as recommended by the manufacturer and tested. Plants using this valve (modified or unmodified) should record each valve actuation and each valve failure. Failures must be reported to the Nuclear Regulatory Commission. The licensee must compare such failure with those of Copes-Vulcan valves with a view toward further modification or replacement, as necessary.

Finding

Power-operated relief valves in PWRs which fail in the open position could contribute to the probability of a small break LOCA.

Recommendations

(e) All PWR licensees should provide a system which uses the block valve to protect against a small break LOCA. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened, to relieve excess pressure. An override feature should be incorporated. Justification should be provided to assure that failure of this system would not decrease overall safety by intensifying plant transients and accidents.

(f) Each licensee should perform a confirmatory test of the automatic block valve closure system installed in response to (a) above.

(g) Westinghouse should submit a report for staff review documenting the various actions which have been taken to decrease the probability of a small break LOCA caused by a stuck-open PORV and show how they constitute sufficient improvements in reactor safety. This report should be submitted for staff review.

(h) Future failures of a PORV to close should be reported promptly to the NRC. Future challenges to the PORVs should be reported in the annual report.

Finding

One possible way to completely eliminate the risk associated with the failure of relief valves is to operate the plants with the block valves closed. This mode of operation, however, could result in an increase in the lift frequency of one safety valve. Information relative to the observed failure rate of safety valves has not been made available. Consequently, neither the desirability or acceptability of this mode of operation can be evaluated at this time.

Recommendations

(i) Safety valve failure rate based on past history of the Westinghouse-designed operating plants should be included in the report mentioned in above.

(j) Future failures of a safety valve to close should be reported promptly to the NRC. Future challenges to the safety valves should be documented in the annual report.

(k) The staff's implementation of the Lessons Learned Task Force long-term recommendations should pursue the interrelationship of safety and relief valves in its future study dedicated to safety and non-safety grade systems (Recommendations No. 9 of NUREG-0585). This study should include an evaluation of the elimination of the PORV function.

3.2.5 Audit Calculations

The following conclusions and recommendations (details may be found in Section 4.2.5 of Appendix VIII.) are based on our audit of the analyses performed by W:

Findings

(a) The calculated system response to the three break sizes analyzed demonstrates the ability of the computer program WFLASH to predict the expected behavior of a depressurization, a pressure hang-up, and a re-pressurization transient. Reasonable assurance is therefore provided that the calculated system response using WFLASH may be used as a base for guidelines in the development of operator training and plant emergency procedures to be used to detect and to mitigate the consequences of a small break LOCA.

(b) The core uncovering and subsequent heatup calculations performed with the RELAP4 computer program are unrealistic, and could result in a non-conservative evaluation of the fuel cladding response. The treatment of steam superheat and steam generation rates in RELAP4 could result in a peak cladding temperature calculation which is several hundred degrees too low.

(c) The model of the steam generator secondary system is important for the evaluation of transients in which auxiliary feedwater is not considered.

Recommendations

(a) Appropriate changes to the RELAP4 computer code program, and to the modeling procedures employed, should be made to eliminate the uncertainties in the heatup calculation.

(b) The effects of accumulator injection on the transient performed with the RELAP4 computer program should be further investigated to determine the amount of condensation realistically expected, and to determine the effect on heat-up and core uncovering.

(c) Since the model of the steam generator secondary system is important for evaluating transients in which auxiliary feedwater is not considered, appropriate changes to the RELAP4 computer program, and to the modeling procedures employed, should be made to more realistically represent the steam generator's behavior.

3.3 Operator Training

Operator training has evolved over the last ten to fifteen years from concentrated on-the-job training programs, with little time allotted to formal training, to the more formal, Commission-approved programs of today. In addition, the expanded use of simulators has contributed significantly to the quality of operator training.

In the past, training programs have underemphasized nonstandard, passive conditions such as misaligned systems, undetected failures of engineered safety features (ESF) equipment and multiple failures. Regardless of the merits of the single failure criterion as a design basis, it should not be considered as a limiting basis for training purposes.

Our review of operator training is discussed in Appendix IX. Significant findings and recommendations are summarized below.

3.3.1 Expanded Use of Simulators in Operator Training

Findings

It is generally acknowledged by the NRC staff and by the operators themselves that simulator operation is a valuable part of operator training. This concensus is reinforced in EPRI Report No. NP-309, which reads in part, "Operations regard simulators as the best vehicle for obtaining operational training . . . it helps you to see casualty modes." It is also apparent from the TMI-2 accident that transient recognition by the operator, and the operator's response based upon his understanding of the plant status, are essential to reactor safety. We believe that a primary part of operator training in event recognition and response should be actual "hands-on" operation in response to various plant transients and accidents. This sort of experience can be gained, to some degree, through actual plant operation and walk-throughs, but must include event simulation and actual operator response and observation to be most meaningful.

Recommendations

(a) All licensed operators be required to participate in a simulator training program to observe such events as a stuck-open power-operated relief valve (PORV) and natural circulation. Training on protecting the cores should be emphasized on all plants. This includes the means to recognize that an adequate heat sink, primary system inventory, and intact primary and secondary systems exists. Simulator training programs should be reviewed to assure that they include the operator errors and equipment failures that contributed to the TMI-2 accident. An evaluation of the simulator control board design and simulated response as compared to the operator's individual response and actual control board design must be made on a case-by-case basis. The differences which may exist must be addressed as part of the operator's training so that negative training feedback will not result.

(b) Plant simulators should offer, as a minimum, the following small break LOCA scenarios:

(a) Continuous depressurization.
(b) Pressure stablized at a value close to secondary system pressure.
(c) Repressurization.
(d) Stuck-open PORV.
(e) Stuck-open letdown valve.

Each of these cases should be simulated with the reactor coolant pumps running and with the pumps not running. The first three events listed above should be simulated for breaks in the hot leg and in the cold leg. In addition to the usual single failures assumed in the ECCS and feedwater systems, the extended (main and auxiliary) loss of all feedwater should be considered (see Section 4.2.2.1 of Appendix VIII).

3.4 Operating Procedures

Operating and emergency procedures are developed in accordance with Regulatory Guide 1.33 Appendix A, "Quality Assurance Program Requirements (Operation)" and Sections 5.3.2 and 5.3.9 of ANSI 18.7 and ANS 3.2, entitled "Administrative Controls and Quality Assurance of Operation of Nuclear Power Plants"

Each normal operating procedure involves the use of checklists, and is based on a controlled evaluation, giving final conditions as goals to achieve. On the other hand, abnormal and emergency procedures are completely different in that the operator is now confronted with automatic responses for which he may have to take manual actions. Therefore, when writing the abnormal and emergency procedures, consideration should be given to the real time that it takes for systems to respond and for the operator to perform a manual function. (The preceding material was previously stated in NUREG-0560.)

As discussed in Appendix IX, the staff requested the plant emergency procedures for loss of coolant, steam line break, loss of offsite power and loss of feedwater events from all operating reactor licensees. Most licensees with W-designed operating plants complied with this request. A review of emergency procedures for ten W-supplied power reactor facilities (some prior to the TMI-2 accident, some after) indicated deficiencies in providing specific operator guidance to monitor, interpret, and respond to critical plant conditions. In general, the procedures failed to guide the operator to monitor and interpret available instrumentation to verify that: (1) reactor coolant system inventory is being maintained, (2) the core has adequate flow for heat removal, and (3) a heat sink is available and operating, therefore assuring the capability for heat removal from the reactor coolant system. For example, the emergency procedures for loss of LOCAs (prior to TMI-2) state that decreasing pressurizer pressure and level are indicative of a LOCA. As emphasized at TMI-2, this is clearly not the case for breaks (or stuck-open valves) in the pressurizer steam space, for which pressurizer level will not be a valid representation of reactor coolant system inventory. IE bulletins issued have directed licensees to take action on these and other areas. Licensee responses to these bulletins are currently being evaluated and separate reports are being issued containing the staff's evaluation. Licensees have in general revised, or are revising, procedures as a result of these IE bulletins.

The Westinghouse Operating Plants Owners Group, in conjunction with Westinghouse, has developed generic guidelines for emergency procedures regarding small break LOCAs. We have reviewed the proposed guidelines and found them acceptable for incorporation into the plant procedures by each licensee (see Appendix IX for details of our review). These approved guidelines contain an acceptable set of criteria for termination of high pressure injection and acceptable criteria for reactor coolant pump trip.

Finding

Procedures are important to the operator training and decision making process during the course of a transient or accident.

Before the TMI-2 accident, little attention was paid to operational procedures, especially to emergency procedures and their relationship to the supporting safety analyses. Unambiguous diagnostics and proper precautions and prohibitions were not always considered in the development of procedures.

Recommendation

The NRC should become more involved in the review of procedures, including their correlation with the assumptions made in the supporting safety analyses. The procedures should include recognitions of the event, precautions, actions, and prohibited actions.

Finding

The NSSS vendor usually does not check a customer's plant operating procedures to determine whether the vendor's operational guidelines have been properly incorporated into the plant procedures.

Recommendation

Independent of the NRC review of procedures, the NSSS vendor should confirm that the vendor's operational guidelines have been properly incorporated into the customer licensee's plant operating procedures. Any exceptions which the customer-licensee may have taken to the vendor's guidelines should be documented with appropriate justification. Copies of the correspondence regarding such exceptions should be transmitted to the NRC for information.

3.4.3 Symptom-Based Emergency Procedures

Finding

Emergency operating procedures currently in use at operating plants have evolved on an "event-specific" basis. Symptom-based emergency procedures, which are categorized according to general plant symptoms and include the essential features of several separate exising procedures, could make use of the fact that the initial operator responses to the associated events are similar.

The principal advantages of the symptom-based emergency procedures over the event-specific procedures are: (1) the procedures as a whole would be simplified significantly and would, therefore, not require the operator to make a detailed diagnosis of the plant conditions prior to consulting an emergency procedure, (2) the total number of emergency procedures with which the operator would have to contend during an emergency would be reduced significantly, and (3) such an exercise would necessitate that the licensees look again at their emergency procedures in a more integrated manner.

We believe that the aforementioned advantages would contribute significantly to the operator's ability to maintain the plant in a safe condition in the event of an emergency

Recommendation

Licensees whose emergency procedures have been developed on an event-specific basis should restructure and reformat them on a symptom basis.

3.5 Human Factors

3.5.1 Monitoring Control Board and Other Instruments

Finding (From NUREG-0560)

The operator must understand his responsibilities during abnormal and emergency conditions. The design basis for the plant has provided that, in the event of emergencies, suitable actions will be automatically initiated by the safety systems. The operator's initial responsibility is to monitor the parameters of interest and verify that appropriate safety systems have been actuated. If the appropriate actuations have not occurred, the operator must intercede and perform the actions necessary to implement them. The operator is trained to believe his instrumentation. However, he must be trained not to rely on a single instrument, since any single indication may be erroneous or misleading under certain conditions. The reason for this precaution was clearly illustrated at TMI-2, where operator attention was focused on the pressurizer level indication. In virtually all situations, other instrumentation can be used to corroborate or refute the validity of a given instrument.

Recommendation (From NUREG-0560)

The operator should monitor the control board and evaluate all parameters of concern by appropriate checking of other instrumentation. He must perform this cross-check to verify instrument display. If he has additional manual actions to perform, he may reduce his observations on other system parameters that may lead him to tunnel vision. This recommendation should be implemented in operator training programs.

TABLE 3-1

SCHEDULE FOR IMPLEMENTATION OF
BULLETINS & ORDERS TASK FORCE RECOMMENDATIONS
FOR W-DESIGNED OPERATING PLANTS

Section No.	Title of Recommendation	Schedule
3.1.3.1	Technical Specification (TS) Time Limit on AFW System Train Outage (GS-1)	1/1/80
3.1.3.2	TS Administration Control on Manual AFW System Valves Lock and Verify Position (GS-2)	1/1/80
3.1.3.3	AFW System Flow Throttling-Water Hammer (GS-3)	1/1/80
3.1.3.4	Emergency Procedures for Initiating Backup Water Supplies (GS-4)	1/1/80
3.1.3.5	Emergency Procedures for Initiating AFW Flow Following Loss of All AC Power (GS-5)	1/1/80
3.1.3.6	AFW System Flow Path Verification (GS-6)	1/1/80
3.1.3.7	Non-safety Grade Non-redundant AFW System Automatic Initiation Signals (GS-7)	1/1/80
3.1.3.8	Automatic Initiation of AFW Systems (GS-8)	1/1/80
3.1.4.1	Primary AFW Source Low Level Alarm	1/1/80
3.1.4.2	AFW Pump Endurance Test	1/1/80
3.1.4.3	Indication of AFW Flow to the Steam Generators	1/1/80*
3.1.4.4	AFW System Availability During Periodic Surveillance Testing	1/1/80
3.1.5.1	Automatic Initiation of AFW Systems (GL-1)	1/1/81
3.1.5.2	Single Valves in the AFW System Flow Path (GL-2)	1/1/81

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*Implementation modified by letter, H. Denton to all operating reactors dated October 30, 1979.

TABLE 3-1 (Continued)

Section No.	Title of Recommendation	Schedule
3.1.5.3	Elimination of AFW System AC Dependency on AC Power Following a Complete Loss of AC Power (GL-3)	1/1/81
3.1.5.4	Prevention of Multiple Pump Damage Due to Loss of Suction Resulting from Natural Phenomena (GL-4)	1/1/81
3.1.5.5	Non-Safety Grade, Non-redundant AFW System Automatic Initiation Signals (GL-5)	1/1/81
3.2.1	Small Break LOCA Analysis	
(a)	Analysis Methods-Appendix K	7/1/80
(b)	Plant-Specific Appendix K Calculations	1/1/81
3.2.2	Role of Non-Safety Equipment in Mitigating Small Break LOCAs	
(a)	Reactor Coolant Pump Trip	1/1/81
(b)	Interaction of Safety and Non-Safety Systems	TMI-2 Action Plan
3.2.3	Michelson's Concerns	
(a)	Two-phase Natural Circulation Experiments	1/1/81
(b)	Instrumentation to Verify Natural Circulation	4/1/80
3.2.4	PORV Failures in W-Designed Plants	
(a)	Confirmation of Anticipatory trip	4/1/80
(b)	PID Controller Modification	4/1/80
(c)	Proposed Anticipatory Trip Modification	Plant-specific
(d)	CCI-Supplied PORV	Plant-specific
(e)	Installation of Auto Isolation of PORVs	7/1/80

TABLE 3-1 (Continued)

Section No.	Title of Recommendation	Schedule
(f)	Testing of Auto Isolation of PORVs	During first refueling outage following installation.
(g)	Westinghouse Report on PORV Failure Reduction	10/1/80
(h)	Reporting PORV Failures and Challenges	Failures: Promptly to NRC Challenges: In Annual Report
(i)	Safety Valve Failure Rate Based on Operational Experience	10/1/80
(j)	Reporting Safety Valve Failures and Challenges	Failures: Promptly to NRC Challenges: In Annual Report
(k)	Evaluate Elimination of PORV Function	TMI-2 Action Plan
3.2.5	Audit Calculations	
	(a) Modifications to RELAP4 Heatup Calculation	NRC Action
	(b) Effects of Accumulator Injection on RELAP4 Calculations	NRC Action
	(c) Modification of RELAP4 to Represent Steam Generator Realistically	NRC Action
3.3.1	Expanded Use of Simulators in Operator Training	
	(a) Simulator Training Program	7/1/80
	(b) Simulation of Small Break LOCA	1/1/81

TABLE 3-1 (Continued)

Section No.	Title of Recommendation	Schedule
3.4.1	Review of Procedures (NRC)	TMI-2 Action Plan
3.4.2	Review of Procedures (NSSS Vendors)	TMI-2 Action Plan
3.4.3	Symptom-Based Emergency Procedures	NRC Action
3.5.1	Monitoring Control Board	4/1/80

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APPENDIX I

COMPARISON OF WESTINGHOUSE-DESIGNED OPERATING REACTORS

1. GENERAL FEATURES

This appendix compares the more salient features of operating nuclear power facilities that use a nuclear steam supply system (NSSS) designed by Westinghouse (W). The information contained herein was obtained from several sources, including W, licensees, and relevant Final Safety Analysis Reports. The information compiled in the tables of this appendix serves as a reference source for much of this report. In addition, Table I-3 compares thermal-hydraulic parameters of W-designed operating plants with typical Combustion Engineering (CE) -and Babcock & Wilcox (B&W)-designed plants.

At present, there are 25 operating nuclear power units in the United States using a W-designed NSSS. These units are listed in Table I-2. Table I-2 compares core thermal power, reactor coolant system volumes, emergency core cooling system (ECCS), high-pressure ECCS pump flow and shutoff head, ECCS initiation setpoints, and, if the high-pressure ECCS pumps do not develop sufficient head to lift the power operated relief valves (PORVs), the capacity of the charging pumps.

W-designed operating reactors can be placed in various categories for purposes of comparison. One method of categorizing these W-designed plants is by the number of reactor coolant loops in a plant. A W-designed plant may have two loops, three loops, or four loops in the primary coolant system (see Figures I-2, I-3, and I-4). The major differences among these three types of plants, as identified by W, are in the reactor core.

1.1 Reactor Coolant Pumps

The reactor coolant pumps used in the facilities listed in Table I-2 vary in horsepower, but W indicates that, except for Yankee Rowe, the design of the reactor coolant pumps is basically the same on all W-designed operating plants. W uses the 93 series pumps on two-loop plants and the 93A and 93 series pumps on three- and four-loop plants.

1.2 Steam Generators

Three different steam generator designs have been used for the W-designed nuclear steam supply systems. The three oldest plants have the 27 series steam generators. The next eight plants have the 44 series steam generators. The 14 newest units have the 51 series steam generators. Table I-1 summarizes the primary and secondary side volumes and heat transfer areas of the steam generators used at the W-designed plants. Estimated boiloff times for W steam generators are given in Table I-2. Appendix II provides a discussion of the parameters affecting dryout times.

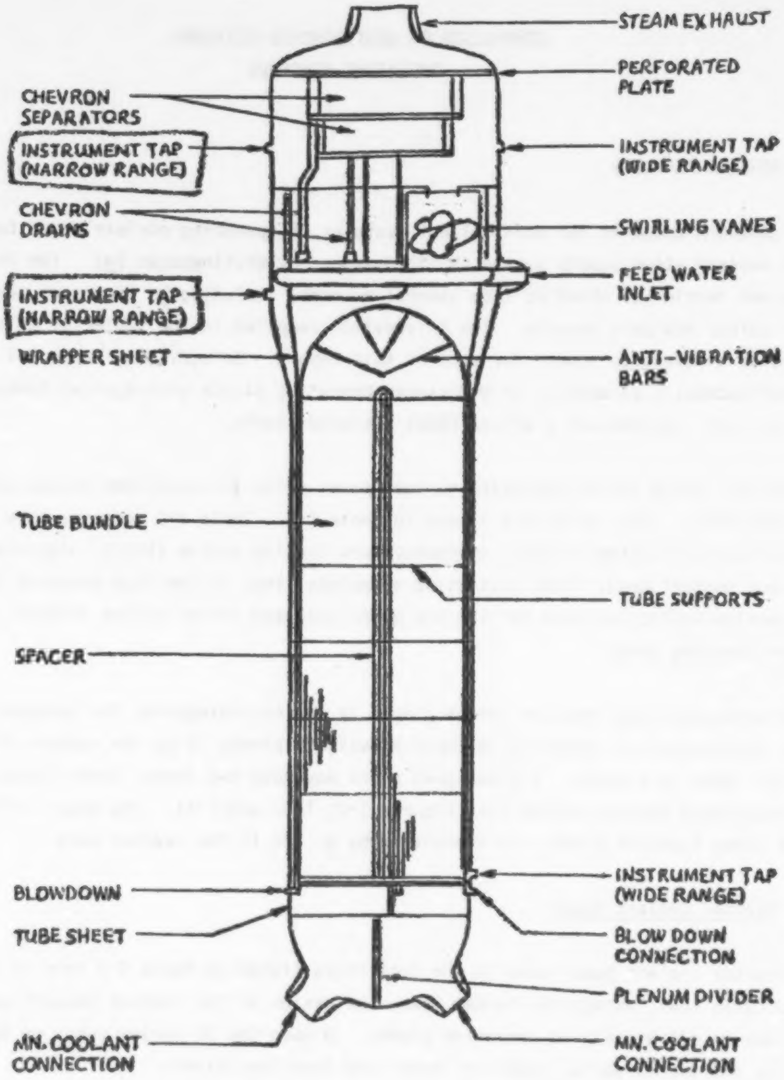
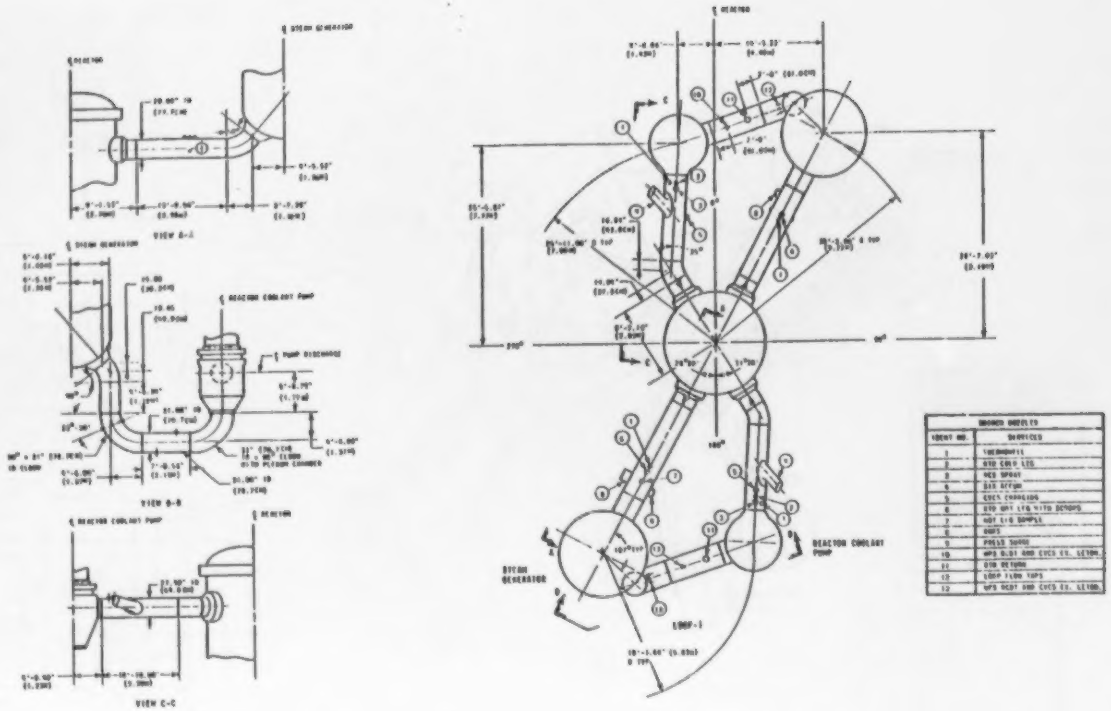

STEAM EXHAUST
PERFORATED PLATE
CHEVRON SEPARATORS
INSTRUMENT TAP (NARROW RANGE)
INSTRUMENT TAP (WIDE RANGE)
CHEVRON DRAINS
SWIRLING VANES
FEED WATER INLET
INSTRUMENT TAP (NARROW RANGE)
WRAPPER SHEET
ANTI-VIBRATION BARS
TUBE BUNDLE
TUBE SUPPORTS
SPACER
BLOWDOWN
INSTRUMENT TAP (WIDE RANGE)
TUBE SHEET
BLOW DOWN CONNECTION
PLENUM DIVIDER
MN. COOLANT CONNECTION
MN. COOLANT CONNECTION

Figure I-1. Typical Westinghouse Steam Generator.



BRANCH NOZZLES	
IDENT NO.	SERVICES
1	THERMOWELL
2	RTD COLD LEG
3	RCS SPRAY
4	SIS ACCUM
5	CVCS CHARGING
6	RTD HOT LEG WITH SCOOPS
7	HOT LEG SAMPLE
8	RHRS
9	PRESS SURGE
10	[illegible] AND CVCS EX. LETDN.
11	RTD RETURN
12	LOOP FLOW TAPS
13	[illegible] AND CVCS EX. LETDN.

VIEW A-A
VIEW B-B
VIEW C-C
STEAM GENERATOR
REACTOR COOLANT PUMP
LOOP-1

Figure I-2. Primary Loop Layout for Standard Westinghouse Two-loop Plant.

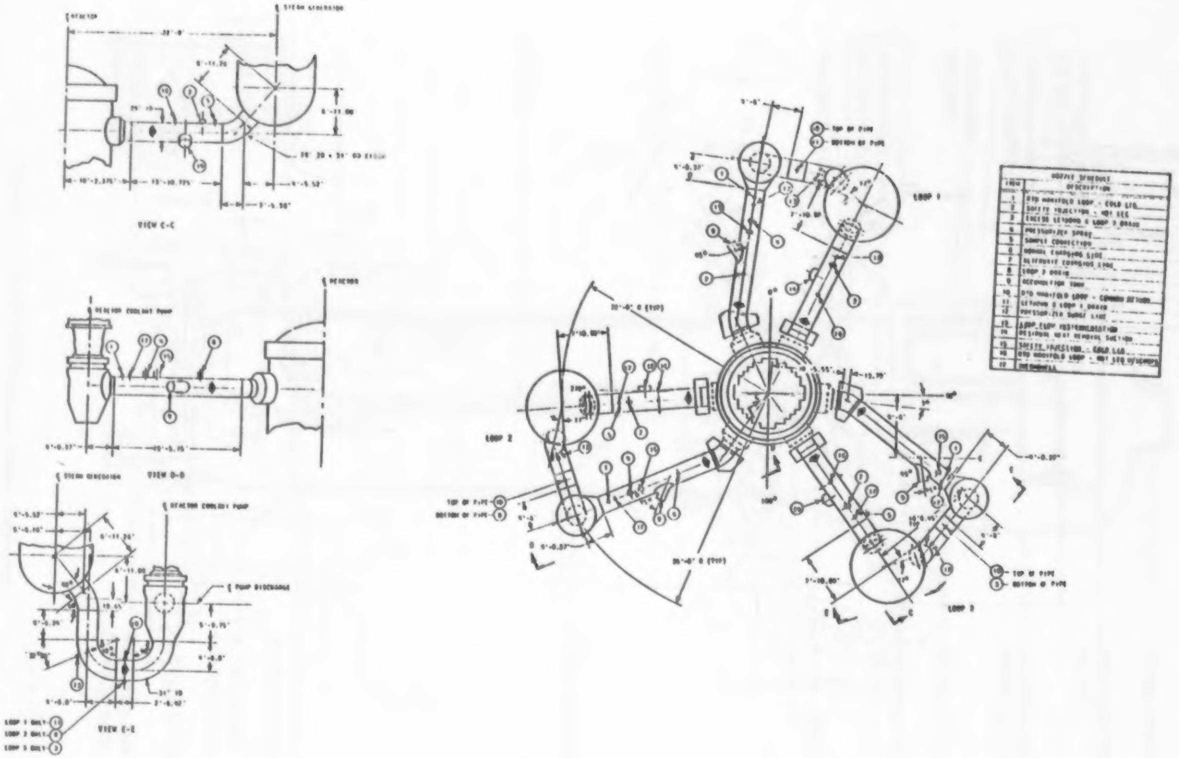
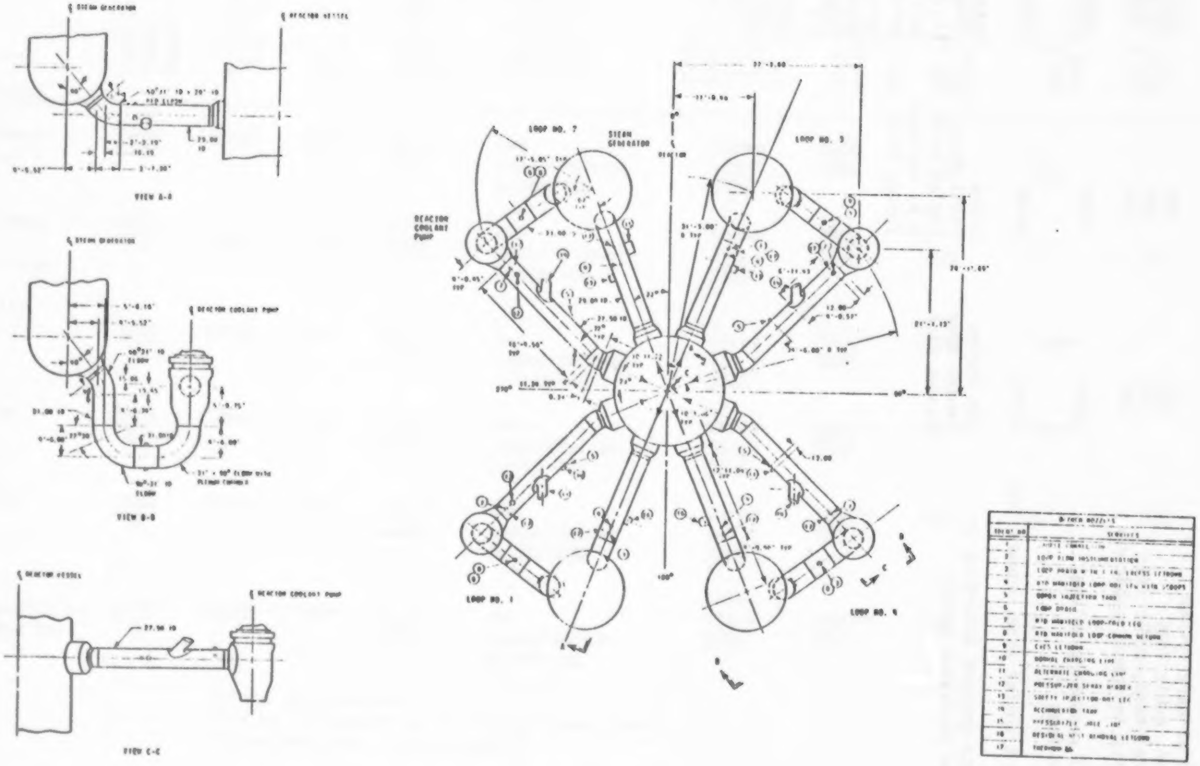

NOZZLE SCHEDULE
DESCRIPTION
LOOP 1
LOOP 2
LOOP 3
TOP OF PIPE
BOTTOM OF PIPE
VIEW C-C
VIEW D-D
VIEW E-E

Figure I-3. Primary Loop Layout for Standard Westinghouse Three-loop Plant Without Stop Valves.


STEAM GENERATOR
REACTOR VESSEL
VIEW A-A
STEAM GENERATOR
REACTOR COOLANT PUMP
VIEW B-B
REACTOR VESSEL
REACTOR COOLANT PUMP
VIEW C-C
LOOP NO. 2
STEAM GENERATOR
LOOP NO. 3
REACTOR COOLANT PUMP
LOOP NO. 1
LOOP NO. 4
0°
90°
180°
270°

BRANCH NOZZLES	
POINT NO	SERVICE
1	[illegible]
2	LOOP FLOW INSTRUMENTATION
3	LOOP DRAIN W TO 1 IN. EXCESS LETDOWN
4	RTD MANIFOLD LOOP HOT LEG WITH SCOOPS
5	BORON INJECTION TANK
6	LOOP DRAIN
7	RTD MANIFOLD LOOP-COLD LEG
8	RTD MANIFOLD LOOP COMMON RETURN
9	CVCS LETDOWN
10	NORMAL CHARGING LINE
11	ALTERNATE CHARGING LINE
12	PRESSURIZER SPRAY NOZZLE
13	SAFETY INJECTION-HOT LEG
14	ACCUMULATOR TANK
15	PRESSURIZER SURGE LINE
16	RESIDUAL HEAT REMOVAL LETDOWN
17	THERMOWELL

Figure I-4. Primary Loop Layout for Standard Westinghouse Four-loop Plant.

TABLE I-1

WESTINGHOUSE STEAM GENERATORS

		27 Series	44 Series	51 Series
1.	Total primary system volume, cubic feet	553	944	1080
	Channel head, cubic feet	206	268	314
	Balance in tubes, cubic feet	347	676	766
2.	Total secondary volume, cubic feet	2592	4580	5868
3.	Heat transfer area, square feet	27700	44430	51500
4.	Plants grouped according to steam generator models	San Onofre 1 Yankee Rowe Haddam Neck	Ginna Point Beach 1&2 Indian Point 2&3 H.B. Robinson Turkey Point 3&4	Kewaunee Salem 1 Trojan Surry 1&2 North Anna 1 Farley 1 Beaver Valley Zion 1&2 D.C. Cook 1&2 Prairie Island 1&2

TABLE I-2 COMPARISON OF KEY OPERATING CHARACTERISTICS OF W-DESIGNED OPERATING PLANTS

Utility/Plant	Core Thermal Power, MWt/ Number Loops	RCS Volume, ft^3 (Inc. Pzr.)	Vol. of Pressurizer, ft^3	PORV Capacity, lb/hr/MWt/set-point, psig	Number of PORVs, Manufacturer Model No.	Number of Safety Valves, Total Capacity, lb/hr/MWt/ Set Point, psig	HPI PUMP CHARACTERISTICS Shut Off Head, ft/psig	gpm @ 1000 psig	gpm @ 1600 psig	gpm @ Safety PORV Set Pt.	Positive Displacement Changing Pump Capacity, gpm
Alabama Power Co., Farley 1	$\frac{2652}{3}$	9723	1400	$\frac{79.2}{2335}$	two Copes-Vulcan D-100-160	three $\frac{130.1}{2485}$	$\frac{6000}{2600}$	750	550	-	$\frac{250}{325}$
Carolina Power & Light, H. B. Robinson	$\frac{2200}{3}$	9343	1300	$\frac{95.5}{2335}$	two Copes-Vulcan D-100-160	three $\frac{130.9}{2485}$	$\frac{3300}{1430}$	350	0	231	N/A
Commonwealth Edison, Zion 1&2	$\frac{3250}{4}$	12000	1800	$\frac{64.6}{2335}$	two Copes-Vulcan D-100-160	three $\frac{129.2}{2485}$	$\frac{6000^{**}}{2600}$	490**	380**	-	$\frac{165}{230}$
Connecticut Yankee, Haddam Neck	$\frac{1825}{4}$	8800	1300	$\frac{115.1}{2270}$	two Copes-Vulcan D-100-160	three $\frac{160.7}{2485}$	$\frac{6800^{**}}{2948}$	~800**	575**	-	$\frac{325}{380}$
Consolidated Edison, Indian Pt. 2	$\frac{2758}{4}$	12224	1800	$\frac{61.3^{*}}{2335}$	two Copes-Vulcan D-100-160	three $\frac{147.9}{2485}$	$\frac{3500^{**}}{1517}$	485	0	294	N/A
Power Authority of State of New York Indian Pt. 3	$\frac{3025}{4}$	12032	1800	$\frac{61.3^{*}}{2335}$	two Copes-Vulcan D-100-160	three $\frac{138.9}{2485}$	$\frac{3500^{**}}{1517}$	485	0	294	N/A

†Data not provided
*To be increased to 78.7 about June 1979
**Charging pump (centrifugal)

TABLE I-2 (Continued)

Utility/Plant	Core Thermal Power, MWt/ Number Loops	RCS Volume, ft^3 (Inc. Pzr.)	Vol. of Pressurizer, ft^3	PORV Capacity, lb/hr/MWt/set-point, psig	Number of PORVs, Manufacturer Model No.	Number of Safety Valves, Total Capacity, lb/hr/MWt/ Set Point, psig	HPI PUMP CHARACTERISTICS				Positive Displacement Charging Pump Capacity, gpm
							Shut Off Head, ft/psig	gpm @ 1000 psig	gpm @ 1600 psig	gpm @ Safety PORV Set Pt.	
Duquesne Light, Beaver Valley 1	2660/3	9716	1400	79.9/2335	three Masoneilan 38-20771	three 129.7/2485	5850/2536	495	380	-	175/225
Florida Power & Light, Turkey Pt. 3&4	2208/3	9343	1300	95.1/2335	two Copes-Vulcan 5-131642	three 132.8/2485	3500/1517	410	0	231	N/A
Indiana & Mich. Electric, D.C. Cook 1	3250/4	†	†	64.6/2335	three Masoneilan 38-20721	three 129.2/2485	5800/2514	560	400	-	70/225
D.C. Cook 2	3400/4	12612	1800	61.8/2335	three Masoneilan 38-20721	three 123.5/2485	5800/2514	560	400	-	170/225
Northern States Power Prairie Is. 1&2	1650/2	6450	1000	108.5/2335	two Copes-Vulcan D-100-160	two 209.1/2485	5000/2168	760	600	180	N/A
Portland Gen. Electric, Trojan	3411/4	12540	1800	61.6/2350	two Copes-Vulcan D-100-160	three 123.1/2485	600/2600	495	380	-	180/230

†Data not provided

TABLE I-2 (Continued)

							HPI PUMP CHARACTERISTICS				
Utility Plant	Core Thermal Power, MWt/ Number Loops	RCS Volume, ft³ (Inc. Pzr.)	Vol. of Pressurizer, ft³	PORV Capacity, lb/hr/MWt/set-point, psig	Number of PORVs, Manufacturer Model No.	Number of Safety Valves, Total Capacity, lb/hr/MWt/ Set Point, psig	Shut Off Head, ft/psig	gpm @ 1000 psig	gpm @ 1600 psig	gpm @ Safety PORV Set Pt.	Positive Displacement Charging Pump Capacity, gpm
Public Service Electric & Gas, Salem 1	$\frac{3338}{4}$	12811	1800	$\frac{63}{2350}$	two Copes-Vulcan D-100-160	three $\frac{125.8}{2485}$	$\frac{6168}{2670}$	490	380	-	$\frac{150}{210}$
Rochester Gas & Electric, Ginna	$\frac{1520}{2}$	6245	800	$\frac{117.8}{2335}$	two Copes-Vulcan D-100-160	two $\frac{189.5}{2485}$	$\frac{3426}{1485}$	285	0	180	N/A
Southern Calif. Edison, San Onofre 1	$\frac{1347}{3}$	6450	1000	$\frac{80}{2190}$	two ACF Industries 70-18-9 DRTX	Two $\frac{178.2}{2500}$	$\frac{6000}{2600}$	570	460	-	$\frac{165}{300}$
Virginia Elec. Power, Surry 1&2	$\frac{2449}{3}$	9200	1300	$\frac{86}{2335}$	two Copes-Vulcan IA58RGP	three $\frac{120.5}{2360}$	$\frac{6000}{2600}$	520	420	-	$\frac{255}{250}$
North Anna 1	$\frac{2775}{3}$	9957	1400	$\frac{76}{2335}$	two Masoneilan 38-20721	three $\frac{137}{2485}$	$\frac{5900}{2550}$	650	550	-	$\frac{250}{325}$
Wisconsin Electric Power, Point Beach 1&2	$\frac{1518}{2}$	6450	1000	$\frac{117.9}{2335}$	two Copes-Vulcan D-100-160	two $\frac{189.7}{2485}$	$\frac{3550}{1539}$	900	0	180	N/A

TABLE I-2 (Continued)

Utility Plant	Core Thermal Power, MWt/ Number Loops	RCS Volume, ft^3 (Inc. Pzr.)	Vol. of Pressurizer, ft3	PORV Capacity, lb/hr/MWt/set-point, psig	Number of PORVs, Manufacturer Model No.	Number of Safety Valves, Total Capacity, lb/hr/MWt/ Set Point, psig.	HPI PUMP CHARACTERISTICS Shut Off Head, ft/psi	HPI PUMP CHARACTERISTICS gpm @ 1000 psig	HPI PUMP CHARACTERISTICS gpm @ 1600 psig	HPI PUMP CHARACTERISTICS gpm @ Safety PORV Set Pt.	Positive Displacement Charging Pump Capacity, gpm
Wisc. Pub. Serv., Kewaunee	1650/2	6191	1000	106/2335	Two Copes-Vulcan D-100-160	Two 209.1/2485	500/2167	750	500	180	N/A
Yankee Atomic Electric, Yankee Rowe	600/4	2940	295	118/2400	One Dresser 31533 VX	two 153/2485	1950/844	0	0	99	N/A

TABLE I-2 (Continued)

Utility/Plant	Steam Generator Time to Dryout, min.	Per Turbine/ Per Motor Aux Feed Cap., % Of Design	ECCS INITIATION SET POINTS			
			Hi Pressure, psig	Low Pzr. coincident w/ Lo Level	Steam Generator ΔP, psi	Hi Stm. Flow w/ Lo-Lo T_{avg} or Low Stm. Press.
Alabama Power Co., Farley 1	37.3	2.3%/1.2%	5.4	1850 psi, 15% level	100	585 psi +
Carolina Power & Light, H. B. Robinson	32.0	2.3%/1.2%	4.0	1715 psi 6% level	100	++ 543°F 614 psi
Commonwealth Edison, Zion 1&2	45.8	2.8%/1.4%	4.5	1815 +++	100	++ 540 °F 600 psi
Connecticut Yankee, Haddam Neck	22.0	2.4%/N/A	5.0	1700 +++	N/A	N/A
Consolidated Edison, Indian Pt. 2	31.5	2.6%/1.3%	2.0 +++	1700	150	++ 540°F 600 psi
Power Authority of State of New York, Indian Pt. 3	29.2	2.6%/1.3%	2.0	1700 +++	150	++ 540°F 600 psi

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+ Only low steam generator pressure.
++ High steam flow or function of load.
+++ Removed low level coincidence.

TABLE I-2 (Continued)

			ECCS INITIATION SET POINTS			
Utility/Plant	Steam Generator Time to Dryout, Min.	Per Turbine/ Per Motor Aux Feed Cap., % Of Design	Hi Pressure, psi	Low Pzr. coincident w/ Lo Level	Steam Generator ΔP, psi	Hi Stm. Flow w/ Lo-Lo T_{avg} or Low Stm. Press.
Duquesne Light, Beaver Valley 1	37.3	+	1.5	1765 5% level	100	++ 543°F 500 psi
Florida Power & Light, Turkey Pt. 3&4	31.3	2.3%/N/A	4.0	1715	100	++ 543°F 600 psi
Indiana & Mich #1 Electric, D.C. Cook 1&2	45.8	2.7%/1.4	1.1	1815 5% level	100	++ 541°F 600 psi
#2	42.1	2.7%/1.4%	1.1	1900 15% level	100	600+ psi
Northern States Power, Prairie Island 1&2	24.7	1.5%/1.5%	4.0	1815	100	500+ psi
Portland Gen. Electric, Trojan	39.3	2.4%/N/A	5.0	1765 5%	100	++ 553°F 600 psi

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+ Only low steam generator pressure.

++ High steam flow or function of load.

+++ Removed low level coincidence.

TABLE I-2 (Continued)

	Steam Generator Time to Dryout, Min.	Per Turbine/ Per Motor Aux Feed Cap., % Of Design	ECCS INITIATION SET POINTS			
			Hi Pressure, psi	Low Pzr. coincident w/ Lo Level	Steam Generator ΔP psi	Hi Stm. Flow w/ Lo-Lo T_{avg} or Low Stm. Press.
Public Service Electric & Gas, Salem 1	43.1	$\frac{2.4\%}{1.2\%}$	4.7	1765 5% level	100	++ 543°F 500 psig
Rochester Gas & Electric, Ginna	29.0	$\frac{2.9\%}{1.4\%}$	6.0	1715 5% level	N/A	500 psi+
Southern Calif. Edison, San Onofre 1	22.5	$\frac{2.1\%}{1.7\%}$	N/A	1685	N/A	N/A
Virginia Electric Power, Surry 1&2	43.1	$\frac{2.7\%}{1.3\%}$	3.0	1715	100	++ 543°F 525 psi
North Anna 1	37.3	$\frac{2.3\%}{1.1\%}$	2.3	1765 5% level	100	++ 543°F 600 psi
Wisconsin Electric Power, Point Beach 1&2	40.0	$\frac{2.6\%}{1.3\%}$	5.0	1735 5% level	N/A	530 psi +

+ Only low steam generator pressure.

++ High steam flow or function of load.

+++ Removed low level coincidence.

TABLE I-2 (Continued)

ECCS INITIATION SET POINTS

Utility/Plant						
Wisc. Public Serv., Kewaunee	44.2	1.2% 1.2%	4.0	1815	N/A	N/A
Yankee Atomic Electric, Yankee Rowe	51.6	1.9% N/A	5.0	N/A	N/A	N/A

+ Only low steam generator pressure.
++ High steam flow or function of load.
+++ Removed low level coincidence.

TABLE I-3

COMPARISON OF THERMAL-HYDRAULIC PARAMETERS

Vendor	Westinghouse										
Reactor	Trojan	Salem 1	Surry 1&2	Farley 1	Turkey Point 3&4	North Anna 1&2	Prairie Island 1&2	Ginna	Point Beach 1&2	H. B. Robinson	Zion 1&2
Design Power, MWt	3411	3338	2449	2652	2200	2775	1650	1300	1519	2192	3250
T_{in}, °F	552.5	544.4	*	543.4	546.2	546.8	535.5	551.9	552.5	546.2	530.2
T_{out}, Core, °F	619.4	611.9	*	613.6	604.5	614.6	601.9	603.9	612.5	604.5	597.0
T_{out}, Vessel, °F	616.7	609.1	*	610.8	602.1	613.8	599.1	601.4	610.1	602.1	594.3
Core Pressure, Psia	2250	2250	2250	2250	2250	2250	2250	2250	2250	2250	2250
Core Flow, 10^6 lb/hr	126.7	126.4	*	96.2	97.0	100.5	65.2	64.3	63.6	97.0	128.9
Core Flow Area, ft²	51.1	51.1	*	41.5	41.8	41.5	27.0	27.0	27.0	41.8	51.4
High Pressure Injection Initiation Setpoint, psia	1765	1765	1715	1850	1715	1765	1815	1715	1735	1715	1815
Average Coolant Subcooling at Injection Pressure, °F	34.0	39.0	*	46.3	37.7	36.4	52.3	35.2	325	40.4	57.4
Subcooling at Core Outlet (normal), °F	33.5	41.1	*	39.4	48.5	38.4	51.1	49.1	40.5	48.5	56.0

*Information unavailable at time of report

**Unit 1/Unit 2

TABLE I-3 (continued)

Vendor	Westinghouse								B&W	CE
Reactor	Haddam Neck	Indian Point 2	Indian Point 3	Beaver Valley 1	D.C. Cook 1&2	San Onofre 1	Kewaunee	Yankee Rowe	TMI-2	Calvert Cliffs 1&2
Design Power, MWt	1825	2758	3025	2652	3250	1347	1650	600	2772	2560
T_{in}, °F	*	543.0	542.6	542.5	536.3	*	535.5	*	557.0	543.4
T_{out}, Core, °F	*	598.5	602.9	611.8	602.0	*	601.9	*	610.6	597.4
T_{out}, Vessel, °F	*	596.0	600.4	609.9	599.3	*	599.1	*	607.7	595.4
Core Pressure Psia	2250	2250	2250	2250	2250	2250	2250	2250	2200	2250
Core Flow 10^6 lb/hr	*	130.0	130.1	96.3	129.5	*	65.2	*	137.8	117.5
Core Flow Area, ft²	*	51.4		41.5	51.4	*	27.0	*	49.2	53.5
High Pressure Injection Initiation Setpoint, psia	1700	1700	1700	1765	1815**/1900	1685	1815	N/A	1615	1578
Average Coolant Subcooling at Injection Pressure, °F	*	42.4	40.4	40.0	51.9**/59.4	*	52.3	*	24.0	33.8
Subcooling at Core Outlet (normal), °F	*	54.5	50.1	41.2	51.0	*	51.1	*	39.0	55.5

*Information unavailable at time of report
**Unit 1/Unit 2

1.3 ECCS and Other Related Considerations

The ECCS high pressure pump data, ECCS initiation setpoints, pressurizer PORV and safety valve capacities and setpoints, and positive displacement charging pump capacities (for cases where the high head pumps cannot lift the safety valves) are provided in Table I-2.

The only portions of the ECCS for operating W-designed plants reviewed for this report were the ECCS pump capacities and, if required, positive displacement charging pump capacities. This review did not attempt to determine if the ECCS designs of all plants satisfy current NRC regulations.

2. TWO-LOOP UNITS

There are six operating W-designed two-loop nuclear units in the U.S. with core thermal power ratings which range from 1518 to 1650 MWt. The two-loop units are Prairie Island 1&2, Ginna, Point Beach 1&2, and Kewaunee. Figure I-2 shows the primary system layout for a two-loop plant. The typical elevation difference on W-designed two loop plants between the top of the active core and the bottom of the steam generator tube support plate is approximately 18 feet. Two-loop plants have 121 fuel assemblies in their cores and use series 93 reactor coolant pumps. See Figure I-1 for a typical steam generator schematic and Table I-1 for steam generator design data.

None of the two-loop units has high head pumps capable of achieving the actuation pressures for the pressurizer PORVs or the safety valves. Each of the units has positive displacement charging pumps capable of delivering a total flow of 180 gpm at pressurizer safety valve relief pressures. Westinghouse indicates that, if the steam generators are assumed to dry out in 25 minutes following loss of all feedwater, then at approximately one hour, the core will begin to uncover, if it is assumed that no ECCS injection occurs and a PORV sticks in the open position. The total positive displacement charging pump flow for each of the two-loop plants meets or exceeds the decay heat removal requirements at one hour, although additional flow may be required, depending on the break size and location.

3. THREE-LOOP UNITS

There are ten operating three-loop W-designed nuclear units in the U.S. with core thermal power ratings that range from 2200 to 2775 MWt. An exception is San Onofre 1, which has a rating of 1347 MWt. The three-loop plants are Farley 1, H. B. Robinson, Beaver Valley 1, Turkey Point 3&4, San Onofre 1, Surry 1&2, and North Anna 1. Figure I-3 depicts the primary system layout for a three-loop unit. The typical elevation difference on W-designed three-loop plants between the top of the active core and the bottom of the steam generator U-tube support plate is approximately 18 feet. Three-loop plants have 157 fuel assemblies in their cores and use series 93 or 93A reactor coolant pumps. See Figure I-1 for a typical steam generator schematic drawing and Table I-1 for steam generator design data.

All the three-loop plants, except H. B. Robinson and Turkey Point 3&4, have high head centrifugal charging pumps capable of lifting the pressurizer safety valves. High pressure injection capacity is important in the event of loss of all feedwater (normal and emergency). The shutoff head on the H. B. Robinson high-pressure injection pumps is approximately 1430 psig and, on Turkey Point 3&4, approximately 1517 psig. However, H. B. Robinson and Turkey Point 3&4 do have positive displacement charging pumps, which can provide a total of 231 gpm at PORV actuation pressures. H. B. Robinson and Turkey Point 3&4 need to have the capability to provide high pressure safety injection flow in sufficient quantity to match or exceed decay heat boiloff before the core becomes uncovered in the event of loss of all feedwater. W indicates that, if the steam generators are assumed to dry out in 25 minutes following loss of all feedwater, then at approximately one hour, inventory must be added (at a rate of 238 gpm) to the primary system to prevent uncovering of the core if the PORV is stuck open. The charging pumps at H. B. Robinson and Turkey Point 3&4 can provide a total flow of about 230 gpm at relief valve setpoint pressures. Although the concept was not analyzed in this review, all of the W-designed three-loop units have PORVs which could be opened in combination with positive displacement charging pump flow to depressurize the primary system sufficiently so that the high head pumps could deliver the required injection flow to the core.

San Onofre 1 is one of two W-designed plants which has stainless steel clad fuel. No review was made for this report of the impact of this design feature on ECCS performance. As stated previously, no attempt was made to determine whether the ECCS designs of all plants satisfy current NRC regulations.

4. FOUR-LOOP UNITS

There are ten operating W-designed four-loop nuclear units in the United States with core thermal ratings which range from 2758 to 3400 MWt, except for Yankee Rowe (600 MWt) and Haddam Neck (1825 MWt). The four-loop plants are Zion 1&2, Haddam Neck, Indian Point 2&3, D.C. Cook 1&2, Trojan, Salem 1, and Yankee Rowe. Figure I-4 depicts the primary system layout for a four-loop unit. The typical elevation difference on these W-designed four-loop plants between the top of the active core and the bottom of the steam generator tube support plate is 18 feet. Four-loop plants have 193 fuel assemblies in their cores and use 93 or 93A series reactor coolant pumps. See Figure I-4 for a typical steam generator schematic drawing and Table I-1 for steam generator design data.

All of the W-designed operating four-loop plants, except Yankee Rowe and Indian Point 2&3, have high head centrifugal charging pumps capable of lifting the pressurizer safety valves, which is an important design feature in mitigating a total loss of feedwater (normal and emergency). The shutoff heads for the Yankee Rowe and Indian Point 2&3 high pressure injection pumps are approximately 844 psig and 1517 psig, respectively. In addition to their high head pumps, Yankee Rowe and Indian Point 2 & 3 have positive displacement charging pumps, which can provide an additional 99 gpm for Yankee Rowe and 296 gpm for Indian Point 2&3 at PORV actuation pressures. Indian Point 2&3 and Yankee Rowe need to have the capability to supply high pressure safety injection flow to the core following loss of all feedwater to match or exceed decay heat boiloff prior to uncovering of the core. For Indian Point 2&3 and Yankee Rowe, the positive displacement charging pumps deliver adequate flow at 1.5 hours (assumed time of uncovering of the core if steam generator dryout time is 30 minutes) to exceed core boiloff. Following loss of all feedwater, the Indian Point 2 steam generator is predicted to dry out in 37 minutes, that of Indian Point 3 in 29 minutes, and that of Yankee Rowe in 51 minutes. In the event that the primary system inventory cannot be maintained, it is important to determine if the PORVs can be manually opened in conjunction with positive displacement charging pump flow and the system depressurized, so that the high head pumps can deliver the required injection flow.

Yankee Rowe is the other one of the two W-designed plants which has stainless steel clad fuel. The D. C. Cook plant has an ice condenser containment. The effects of these design features on ECCS performance were not evaluated for this report. As stated previously, the subject evaluation did not determine whether the ECCS designs of all plants satisfy current NRC regulations.

5. ECCS CHALLENGES

The frequency of reported challenges to the safety injection system for W-designed reactors is estimated in this section, based on our survey of licensees and operational data. Any event where ECCS initiation setpoints were not actually exceeded, but an ECCS injection occurred anyway, is defined as a spurious ECCS injection event. All other ECCS injections are categorized as required ECCS challenges. W-designed reactors have experienced ECCS initiations at approximately the same frequency as CE-designed reactors (less than one per reactor year) and B&W-designed reactors, except for two per reactor year for the two Zion units. Many of the ECCS initiations which occurred when the ECCS setpoints were exceeded were due either to the secondary system overcooling the primary system or to steam flow problems. The frequency of initiations per reactor year does not appear to be affected by the fact that W uses four different parameters for ECCS initiation rather than two as is usually done by CE and B&W.

Safety analyses regarding the ECCS presented in final safety analysis reports (safety injection systems) calculate that ECCS initiation is required for some transients, which is consistent with the reported ECCS initiations for W-designed units.

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APPENDIX II

MAIN FEEDWATER SYSTEMS IN WESTINGHOUSE-DESIGNED OPERATING REACTORS

1. GENERAL FEATURES

The main feedwater systems among the 25 licensed Westinghouse (W)-designed units (located at 18 sites) are functionally similar. There are minor design differences in the main feedwater systems between units at a multi-nuclear plant site. Twenty-two of the plants use W turbine-generators and two use General Electric turbine-generators (D.C. Cook Unit 2 uses a Brown-Boveri turbine-generator). Since the nuclear plants were designed by ten different engineering firms, there are some design differences, but the functional performance remains the same. Table II-1 provides an indication of the similarities and differences. The loss of feedwater at Three Mile Island Unit 2 (TMI-2) has been attributed to difficulty occurring in the condensate demineralizer within the feedwater system. This section will be limited primarily to a discussion of the condensate demineralizers in the W-designed plants.

2. FULL-FLOW DEMINERALIZER

Of the twenty-five W-designed operating units, seven units use full-flow demineralizers in their main feedwater system design. Normally, all condensate is processed through the full-flow demineralizer. However, on occasion, the pressure losses through the demineralizers become excessive. In such cases, the flow is then bypassed around the demineralizer while the demineralizer is being serviced. The demineralizer bypass for those plants that have incorporated full flow demineralizers into their design is discussed below:

2.1 Prairie Island 1 & 2

The condensate demineralizers are automatically bypassed by an air-operated valve on a high differential pressure across the demineralizers (45 psid). The air-operated valve would fail in the "open" position upon loss of air.

2.2 Trojan

The condensate demineralizers are automatically bypassed by a motor-operated valve on a high differential pressure across the demineralizer (30 psid). The motor-operated valve will fail "as-is" upon a loss of power.

2.3 Salem 1

The condensate demineralizers are bypassed manually by a motor-operated valve. The valve would remain "as-is" on loss of power.

TABLE II-1 MAIN FEEDWATER SYSTEMS

Components		Plant Name					
		Farley 1	H.B. Robinson	Zion 1&2	Haddam Neck	Indian Pt. 2&3	Beaver Valley 1
Pumps:	Type	Centrifugal (2)	Centrifugal (2)	Centrifugal (3)	Centrifugal (2)	Centrifugal (2)	Centrifugal (2)
	Capacity	15,000 gpm @ 5250 RPM	12,690 gpm	Three pumps @ 50% of normal full flow (15,800 gpm @ 1160 psig)	50%-9600 gpm	15,300 gpm @ 970 psig	15,200 gpm @ 1900 ft.
	Shutoff head	3400 ft.	2400 ft.	1600 psig	1100 psig	1830 ft.	1050 psig
Drives:	Type	2 Steam	2 Electric	2 Steam 1 Electric	2 Electric	2 Steam	2 Electric
	Supply/Exhaust	1. Main steam/ 2. Extraction steam/ main condenser	NA	Main steam/ main condenser	NA	Main steam/ main condenser	NA
	Trips	1. Low suction press. 2. Low lube oil press. 3. Low control oil press. 4. Thrust bearing (normal & reverse) 5. Low condenser vacuum 6. Hi SG level (2/3/SG) 7. Hi water level in MS valve room 8. Turbine overspeed 9. Safety injection 10. Manual (local & remote)	1. Low suction press. 2. Low lube oil press. 3. Loss of cond. pump 4. Electrical overload 5. Bus undervoltage 6. Safety actuation signal 7. Minimum flow (blocked for 30 sec.) 8. Hi SG level	1. Generator trip 2. Low lube oil press. 3. Low control oil press. (turbine only) 4. O.S.T (turbine only) 5. Thrust bearing wear (turbine only) 6. High vibration (turbine only) 7. Electrical fault (motor only) 8. Poor condenser vacuum (turbine only) 9. Low suction press. 10. Manual	1. Low suction press. 2. Condensate pump trip-trips "A" feed pump 3. Normal motor protection trips	1. Turbine overspeed 2. Low bearing oil pressure 3. Low condenser vacuum 4. Thrust bearing wear 5. Manual (local & remote 6. Safety injection signal (Unit 2) 7. Main unit trip (Unit 3) 8. Low suction press. (Unit 3) 9. Loss of bearing oil pump (Unit 3)	1. Motor protection trip 2. Safety injection signal 3. Hi-Hi SG Level 4. Low suction pressure 5. Low oil pressure 6. Manual

TABLE II-1 (Continued)

	Plant Name					
Components	Turkey Pt. 3&4	D.C. Cook 1&2	Prairie Island 1&2	Trojan	Salem 1	Ginna
Pumps: Type	Centrifugal (2)	Centrif. (2)	Centrif. (2)	Centrif. (2)	Centrif. (2)	Centrif. (2)
Capacity	60%-13,000 gpm @ 1880 ft.	65-70% full power 16,750 gpm	65% full power 7.1×10^6 lbs/hr	70% full power 19,800 gpm @ 2020 ft.	50% full power 18,613 gpm @ 884 psig	50% full power 14,000 gpm @ 853 psig
Shutoff head	2650 ft.	2625 ft.	1050 psig	3250 ft.	1257 psig	1180 psia
Drives: Type	2 Electric	2 Steam	2 Electric	2 Steam	2 Steam	2 Electric
Supply/Exhaust	NA	Main steam/ main condenser	NA	1. Main steam 2. LP aux steam/ main condenser	1. Main steam 2. Reheat steam 3. Auxiliary steam/ main condenser	NA
Trips	1. Low suction (200 psi) 2. Low lube oil (3 psig) 3. Safety injection signal 4. Hi-Hi SG level	1. Low suction press. 2. Low lube oil press. 3. Overspeed trip 4. Thrust bearing wear	1. Loss condensate pumps 2. Hi SG level (2/3 > 67%) 3. Safety injection signal 4. Bus undervoltage (< 75% of rated for > 5 sec.)	1. Overspeed (110%) 2. Low oil press. 3. Thrust wear wear 4. Exhaust hood hi temp.	1. Low lube oil level 2. Low lube oil press. 3. Thrust bearing 4. Turbine overspeed	1. Loss offsite power 2. Safety injection signal 3. Overcurrent 4. Thermal reload

TABLE II-1 (Continued)

Components	Plant Name					
	Farley 1	H.B. Robinson	Zion 1&2	Haddam Neck	Indian Pt. 2&3	Beaver Valley 1
Condensate Pumps: No./Strainers	3/yes (inlet-outlet)	2/None	4/yes (suction)	2/None	3/No (Unit 3) Yes (Unit 2)	2/none
Demineralizers: No./No. for Full Power/Mfg.	None	None	None	None	None	None
Bypass/Operation/ Fail Position	NA	NA	NA	NA	NA	NA
FW Heaters: Bypass/ Operation/Fail Position	NA	Yes/manual/NA	Yes/manual/NA	NA	Yes/manual/NA	Yes/ADV Auto/ Closed
Booster Pumps: No.	None	None	4	None	None	None
Control Valves: Auto Isolation Capabilities	1. Rx trip & T_{avg} (low 554°F(2/3)) 2. Hi-Hi SG level (2/3/SG) 3. Safety injection signal	Manual closing Block valves	1. Hi SG level (60°NR) 2. Turbine trip coincident with Tavg < 553°F	Air operated control valve Fails closed	1. Safety injection signal 2. Unit trip & T_{avg} > 554°F (Unit 3)	NA

TABLE II-1 (Continued)

Components	Plant Name					
	Turkey Pt. 3&4	D.C. Cook 1&2	Prairie Island 1&2	Trojan	Salem 1	Ginna
Trips (continued)	5. Loss of condensate pump w/o start of 3 pump	5. Abnormal exhaust vacuum	5. Low suction press.	5. Low vacuum	5. Loss of vacuum at turbine exhaust	5. Manual
	6. Std. overcurrent protection	6. Hi SG Water level.	6. Low lube oil	6. Safety injection signal	6. Hi turbine exhaust temp.	
	7. Bus stripping on undervoltage	7. Safety injection signal	7. Manual	7. Hi-Hi SG water level	7. Low feedwater suction press.	
	8. Bus lock out relay	8. Manual		8. Feedwater pump Hi discharge press.	8. Safety injection signal or Hi-Hi or SG level	
	9. Manual			9. Condensate	9. Manual	

TABLE II-1 (Continued)

	Plant Name					
Components	Turkey Pt. 3&4	D.C. Cook 1&2	Prairie Island 1&2	Trojan	Salem 1	Ginna
Condensate Pumps: No./Strainers	2/None	3/None	3/Yes (hotwell suction)	2/yes (suction)	3/yes (suction)	3/None
Demineralizers: No./No. for Full Power/Mfg.	None	None	3/2/DeLaval	8/6/Graver	6/5/Graver	4/3/Cochrane
Bypass/Operation Fail Position	NA	NA	Yes/AOV auto on HiΔP (45 psid)/open	Yes/MOV auto open on HiΔP (30 psid)/as-is	Yes/MOV man./ as-is	Yes/AOV auto mod. to main. ΔP @ 40 psid/as-is
FW Heaters: Bypass/Operation/ Fail (Position)	Yes/manual/closed	Yes/auto on low FW suction/ open	Yes/manual/NA	Not available	Yes/auto/open & auto/ as-is	yes/man/NA
Booster Pumps: No.	3	3	None	None	None	3
Control valves: Auto Isolation	Reg valve serves As auto isolation valve	Not available	Auto closes 1. Hi SG level 2. Safety injection signal 3. Rx trip and low T_{avg} (554°F)	Not available	Valve closes 1. Hi-Hi SG level 2. Safety injection signal	With SG level control in auto mode 1. Primary coolant low T_{avg} 554°F both SG isolated 2. Safety injection both S.G. 3. Hi SG level 68%- affected SG only Manual mode 1. Safety injection signal 2. Hi SG level

TABLE II-1 (Continued)

Components	Plant Name					
	North Anna	Point Beach 1&2	Kewaunee	Yankee Rowe	San Onofre 1	Surry 1&2
Pumps: Type	Centrifugal (3)	Centrifugal (2)	Centrifugal (2)	Centrifugal (3)	Centrifugal (2)	Centrifugal (2)
Capacity	50% capacity 16,250 gpm	50% capacity 780 gpm @ 2170 ft.	60% capacity 10,000 gpm	2160 gpm @ 1280 ft	50% full power 14,000 gpm @ 853 psig	50% full power 13,800 gpm
Shutoff head	2775 ft	2450 ft.	2520 ft.	2110 ft.	1180 psia	1700 ft.
Drives: Type	3 Electric	2 Electric	2 Electric	3 Electric	2 Electric	2 Electric
Supply/Exhaust	NA	NA	NA	NA	NA	NA
Trips	1. Low suction press. 2. Low lube oil press. 3. Bus undervoltage 4. Phase or neutral current 5. Safety injection signal 6. Hi-Hi SG level 7. Motor speed low & phase current 8. Manual	1. Hi SG level 2. Safety injection signal 3. Avg. temp. error signal (t_{avg} & no load t_{ref}) 5. Low suction press (2 min delay) 6. Loss of condensate pumps (two) 7. Hi-Hi SG level in either SG 8. Manual	1. Undervoltage on supply bus 2. Bus lock out 3. Safety injection signal 4. Low lube oil press.	1. Low suction press. 2. Overload/under voltage 3. Rx scram trip 4. Manual	1. Inverse time and instantaneous overcurrent relays 2. Breaker trip initiated by undervoltage relay 3. Manual	1. Motor protect. on pumps 2. Low suction oil press. 3. Low lube oil press. 4. Bus undervoltage

TABLE II-1 (Continued)

Components	Plant Name					
	North Anna 1&2	Point Beach 1&2	Kewaunee	Yankee Rowe	San Onofre 1	Surry 1&2
Condensate Pumps: No./Strainers	3/yes (suction side)	2/None	2/yes (suction)	3/None	4/yes (hotwell outlet)	3/yes (suction)
Demineralizers: No./No. for Full Power/Mfg.	5/4 Graver	None	None	None	None	None (install. planned)
Bypass/Operation/ Fail Position	Yes/(AOV) auto on Hi ΔP Dem/ open	NA	NA	NA	NA	NA
FW Heaters: Bypass/ Operation/Fail Position	Yes/Manual/NA	Yes/Auto/NA	Yes/Manual/NA	Yes/Manual/NA	Yes/Manual/NA	Yes/Manual/NA
Booster Pumps: No.	None	None	None	None	None	None
Control Valves: Auto Isolation	Reg. Valve Auto Closes 1. Low T_{avg} plus Rx trip F.W. Iso. Valves Auto Close 2. ΔP across Cont. Iso. Valve 3. Safety injection signal 4. Hi-Hi S.G. Level	Auto Close 1. Safety injection signal 2. Hi-Hi SG Level	Auto Close 1. Feedwater isolation (Safety injection signal) 2. Hi-Hi SG level 3. Rx trip with low T_{avg}	Air oper, as-is	Auto Close on Safety injection signal	Auto Close 1. Hi-Hi SG level T_{avg} <554°F 2. RX trip

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2.4 Ginna

The flow through the condensate demineralizers is maintained by an automatically modulated, air-operated valve, which maintains the pressure differential across the demineralizers at 40 psid. The air-operated valve would fail "as-is" on loss of air.

2.5 North Anna 1

The condensate demineralizers are automatically bypassed by an air-operated valve on a high differential pressure across the demineralizers. The air-operated valve would fail in the "open" position upon loss of air.

The results of our investigations of W-designed operating plants indicate that failures in the demineralizers do not represent a significant problem that will contribute to the number of loss of feedwater events that occur at a facility. However, only one aspect of the main feedwater system has been addressed above.

Other component failures in the main feedwater system could result in the loss of feedwater event. These include spurious pump trip signals, inadvertent valve closures, loss of condensate pump or booster pumps, clogged strainers, loss of condenser vacuum, loss of circulating water to condenser, and the loss of feedwater heaters with failure to bypass. The loss of feedwater transient at TMI-2, along with other failures, was instrumental in initiating the accident sequence. A loss of feedwater event followed by failure to inject any auxiliary feedwater (AFW) in 20-45 minutes could result in potentially severe consequences. For this reason, we have investigated the AFW system design and recommended changes, where necessary, which would reduce the unavailability of this system (See plant-specific discussion of AFW system in Appendix X).

3. LOSS OF FEEDWATER FLOW

During normal plant operation at power, the heat transferred to the primary coolant by the core is balanced by the heat removal at the steam generators, which are supplied with feedwater by the main feedwater pumps and discharge steam to the main turbine. Any changes in the secondary system which affect the heat removal capability of the steam generators, such as turbine trip or loss of main feedwater flow, will affect the primary system. In the event of a temporary loss of main feedwater flow, a number of corrective actions are taken to keep the primary and secondary system pressures, temperatures, and water inventories within acceptable limits. These include reactor scram, reestablishment of a reduced feedwater flow by the AFW system, and pressure relief and energy removal by power-operated relief valves (PORVs), steam dump valves, and safety valves.

The AFW system can be used to supply feedwater during normal startup, shutdown, and hot standby operations, as well as under emergency conditions, such as that involving loss of main feedwater flow. For operating plants with a W-designed nuclear steam supply system (NSSS), the AFW system typically has two trains, one with a steam turbine pump drive and the other with one or two pumps with electric motor drives. Some units, however, have two trains of turbine-driven pumps. The flow capacity of the AFW system as a percent of normal feedwater flow on a per turbine drive or per electric motor drive basis is given in Table I-2 in Appendix I. The flow capacity for one train, in gallons per minute, ranges from 1.2 to 2.9 percent of the main feedwater flow at full power. Since the AFW has a higher density and lower enthalpy than the main feedwater, these values are equivalent to steam generator heat removal rates ranging from about 1.7 to 4.2 percent of full power and to decay heat production rates at times from about 0.5 to 28 minutes after reactor scram, respectively.

A complete loss of normal feedwater flow could result from (a) loss of all main feedwater pumps, (b) loss of condensate pumps, (c) a control system malfunction causing closure of the feedwater control valves, (d) operator error involving manual closure of the feedwater control valves, (e) operator or procedural error involving manual closure of feedwater suction or discharge valves. Other transients, such as loss of offsite power or loss of condenser vacuum, also involve loss of normal feedwater flow.

4. STEAM GENERATOR DRYOUT TIME

The consequences of a temporary total loss of feedwater flow (main plus auxiliary) and the times available for corrective manual actions are affected, in part, by the steam generator secondary side water inventory and the type of signal used to obtain a reactor scram. The U-tube type of steam generators used in W-designed plants have larger secondary side water inventories that the once-through steam generators used by Babcock & Wilcox (B&W)-designed plants. However, early initiation of reactor scram is still required to give large steam generator dryout times following the assumed total loss of feedwater flow.

If no protective actions (e.g., reactor trip, AFW system initiation) were taken to protect the reactor, the loss of normal feedwater would lead to a rapid reduction in the secondary side water inventory. For plants with a W-designed NSSS, steam generator dryout times without scram and AFW flow range from about 52 to 74 seconds. These values are based on estimated steam generator secondary side water inventories at full power, the latent heat of vaporization at the lowest setpoint of the steam generator safety valves, instantaneous loss of feedwater flow and a constant power input to the steam generators equal to the full power value. If the reactor is scrammed early in the transient and AFW is still not available, the dryout time is increased appreciably. For plants with a W-designed NSSS, loss of normal feedwater flow results in a plant trip due to low steam generator water level coincident with steam/feed flow mismatch. Westinghouse estimates of the steam generator water inventory at the time when the level decreases to the setpoint value are presented in Table II-2, along with estimates of the dryout time if auxiliary feedwater is not initiated. The dryout times include the effects of decay heat, stored energy in the fuel, and the delay from the time when the low steam generator level coincident with steam/feed flow mismatch setpoint is reached until the reactor is tripped. Modified dryout times, which include an approximate correction of 12 full power seconds for the effect of stored energy in the reactor coolant system water and metal, are also provided in Table II-2.

TABLE II-2 STEAM GENERATOR DRYOUT TIMES FOR PLANTS WITH WESTINGHOUSE-DESIGNED NUCLEAR STEAM SUPPLY SYSTEMS

Plant	MWt	Turbine Trip for % Full Power Greater Than:	Trip for Steam/Feedwater Flow Mismatch Coincident with Steam Generator Low Level	Dryout Time, Minutes	Modified Dryout Time,* Minutes
Yankee Rowe	600	8%	No	51.6	40.1
Haddam Neck	1825	10%	Y	22.0	13.5
North Anna 1	2758	10%	Y	37.3	27.5
Indian Point 3	3025	10%	Y	29.2	20.0
D. C. Cook 1	3250	10%	Y	45.8	35.0
Zion 2	3250	10%	Y	45.8	35.0
Zion 1	3250	10%	Y	45.8	35.0
D. C. Cook 2	3403	10%	Y	42.1	32.0
Trojan	3411	10%	Y	39.3	29.2
Salem 1	3350	10%	Y	43.1	43.1
San Onofre	1351	10%	Mismatch Only	22.5	14.0
Robinson 2	2260	10%	Y	32.0	22.5
Turkey Point 3	2208	10%	Y	31.3	22.0
Turkey Point 4	2208	10%	Y	31.3	22.0
Surry 1	2449	10%	Y	43.1	32.5
Surry 2	2449	10%	Y	43.1	32.5
Indian Point 2	2758	10%	Y	31.5	22.1
Farley 1	2660	50%	Y	37.3	27.3
Beaver Valley 1	2660	10%	Y	37.3	27.3
Ginna	1520	50%	Y	29.0	20.0
Prairie Island 1	1650	50%	Y	24.7	16.0
Prairie Island 2	1650	50%	Y	24.7	16.0
Point Beach 1	1518	50%	Y	40.0	29.8
Point Beach 2	1518	50%	Y	40.0	29.8
Kewaunee	1650	10%	Y	44.2	33.5

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*Including approximate correction of 12 full power seconds to account for stored energy in reactor coolant system fluid and metal.

Loss of all normal alternating current (ac) power could be caused by complete loss of the offsite grid together with a turbine-generator trip, or by loss of the onsite normal ac distribution system. The reactor will trip as a result of a (a) turbine trip, (b) low reactor coolant system flow trip, or (c) loss of power to the control rod drive mechanisms. Normal feedwater flow is also lost for this transient. However, the reactor should trip earlier in the transient, and the dryout times in case of failure of AFW flow initiation should be longer than those for the low steam generator level trip presented in Table II-2.

If a loss-of-coolant accident (LOCA) is postulated to occur coincident with total loss of feedwater flow, the steam generator dryout times would be larger since the break flow removes decay heat and reactor scram from a low pressurizer pressure signal could occur earlier in the transient. For very small breaks, (~ 0.02 ft^2 or less), the steam generator would serve to remove most of the decay heat and the increase in dryout time would be relatively small. For large breaks, there would be a rapid depressurization of the reactor coolant system. In this case, the break and emergency core cooling system (ECCS) injection flows provide more than enough cooling to remove stored and decay heat and, in fact, the steam generators tend to act more as a heat source than heat sink during the transient.

5. EXTENDED LOSS OF FEEDWATER WITH NO BREAK

The initial response of the secondary side of the steam generators to a loss of main feedwater is a decrease in subcooling in the lower part of the steam generator, an increase in steam production rate, an increase in steam pressure and secondary side temperatures, and a decrease in secondary side water inventory. The net result is an initial increase in steam pressure and secondary side temperatures, and a decrease in secondary side water inventory. In addition there is an initial increase in reactor coolant temperature and pressure and either a small increase in reactor power (assuming a positive moderator temperature coefficient of reactivity - a conservative assumption) or a decrease in reactor power (assuming a negative moderator coefficient of reactivity - a realistic assumption), until reactor trip results from a low steam generator level signal. For an assumed instantaneous loss of feedwater flow at full power, the reactor trip occurs at about 12 seconds into the transient. Turbine trip resulting from reactor trip causes an additional temporary increase in secondary side pressure and temperature, which would be limited by the safety valves. This increment could eventually be reduced and stabilized by action of the atmospheric dump or steam bypass valves. These secondary side temperature changes resulting from turbine trip could result in further primary side temperature changes which result in an additional increase in primary side pressure to a maximum pressure below the opening set point of the PORVs. The rapid decrease in reactor power to decay heat levels combined with the continued high heat removal rates at the steam generators then results in a decrease in the average reactor coolant temperature, which tends to approach values somewhat above the secondary side temperature. The reactor coolant system pressure also drops because of the temperature decrease, but is maintained well above the secondary side pressure (~2000 psig) by the pressurizer. After several minutes into the transient, system parameter changes are relatively slow, as decay heat and steam generator water inventory decrease.

If AFW flow is supplied early in the transient and in sufficient quantity, the reduction in steam generator water inventory due to boiloff is offset before there is a large reduction in steam generator cooling capability and the reactor coolant system pressure remains below the set point of the PORVs.

The plant could then be brought to a stabilized condition with recovery of steam generator water level and control of steam pressure by the atmospheric dump or bypass valves. However, if feedwater flow (main or auxiliary) remains unavailable, the continued decrease in steam generator secondary side water inventory results in degraded heat transfer and eventual dryout and complete loss of the steam generator heat sink function. As noted above, steam generator dryout times for operating plants with a W-designed NSSS range from approximately 13 to 40 minutes. This reduction and eventual complete loss of heat removal capability produce an increase in reactor coolant temperature with a corresponding decrease in reactor coolant density. The coolant density decrease results in an insurge into the pressurizer and an increase in reactor coolant system pressure to the set point of the PORVs (~2400 psig).

During the first part of this transition to steam generator dryout, the reactor coolant is still subcooled and the surge rate into the pressurizer is relatively small, since it is controlled by decay heat rates. Since steam is discharged through the PORVs at this point in the transient, there is a high volumetric flow rate (based on conditions upstream of the valve) which tends to offset further pressure increase. If the size of the PORVs is sufficient to drop the pressurizer pressure to the closing setpoint, the pressurizer pressure will fluctuate over a range somewhat larger than that corresponding to the differences between the opening and closing setpoints, as the valves open and close intermittently. The pressure fluctuations should not result in significant flashing in the pressurizer. If the valves do not have sufficient to drop the pressurizer pressure to the closing setpoint, capacity discharge, the PORVs remain open and there is no flashing in the pressurizer.

The continued heat addition to the reactor coolant and flow to the pressurizer finally result in a water-solid pressurizer. With water flow through the PORVs, there is a sharp reduction in volumetric flow rate for a given pressure, which results in an increase in pressure to the lowest set point for the safety valves (~2500 psig). The pressure will continue to increase until the volumetric flow rate through the safety and relief valves offsets the insurge into the pressurizer resulting from the decrease in the reactor coolant density. The decrease in decay heat and the increase in volumetric flow rate (for a given upstream pressure), as the flow conditions upstream of the valves change progressively from low quality two-phase flow to steam, cause a reduction in pressure. Eventually, the pressure is reduced to the point where the safety valves have closed and the pressure again fluctuates about the opening and closing set points of the relief valves until the core uncovers. W estimates that uncovering of the core will occur about one hour after the initial loss of feedwater, if no corrective actions are taken. These estimates were based on analyses performed with the WFLASH (1) code. More recent analyses performed by W with the NOTRUMP (2) code (which uses a more detailed model of the steam generator) indicate that the core will not be uncovered during the entire transient.

After steam generator dryout occurs, the driving force for natural circulation, which results from having a heat sink at a higher elevation than the core, is lost. If the reactor coolant pumps are stopped, some natural circulation should still occur for a while, since void production should occur first at the core exit and the hot legs. Flow would also be induced as the result of the removal of reactor coolant through the pressurizer surge line, which is connected to a hot leg. As the reactor coolant system empties, steam will accumulate in the U-tubes of the steam generators in sufficient quantity to prevent natural circulation.

With the exception of H. B. Robinson, Indian Point 2 and 3, Turkey Point 3 and 4, Prairie Island 1 and 2, Ginna, Point Beach 1 and 2, Kewaunee, and Yankee Rowe, operating plants with a W-designed NSSS have high head centrifugal charging pumps (high head safety injection pumps) with dead head pressures ranging from 2514 to 2948 psig. Manual or automatic initiation of the high pressure safety injection system for plants which have high head pumps with shutoff heads in excess of 2500 psig may result in injection of ECCS water and may prevent uncovering of the core. Manual or automatic initiation of high pressure safety injection for the rest of the W-designed NSSS units would not prevent uncovering of the core, unless other actions could be taken to reduce reactor coolant system pressure to below the dead head pressure of the ECCS pumps.

Some reduction in reactor coolant system pressure later in the transient may be obtained by manually opening the PORVs and using the positive displacement charging pumps without letdown. The charging pump flow would also increase the time to uncovering of the core. For those W-designed NSSS units with high head safety injection pump dead head pressure below 2500 psig, the total positive displacement charging pump capacity varies from 99 to 294 gpm, which is equivalent to about 50 lbm/hr-MWt on all plants. An exception is Yankee Rowe, which has an equivalent mass flow of 88 lbm/hr-MWt. The total steam discharge capacities of the PORVs on operating W-designed plants vary from 118 to 236 lbm/hr-MWt.

It appears that Farley 1, Zion 1 and 2, Haddam Neck, Beaver Valley 1, D. C. Cook 1 and 2, Trojan, Salem 1, San Onofre 1, Surry 1 and 2, and North Anna 1 have the capability to withstand extended loss of all feedwater because of the high safety injection pump dead-head pressure.

The above scenario for plant response to extended loss of feedwater flow was based on general considerations of such factors as valve flow rates, decay heat removal rates, and system water inventories. The W Operating Plants Owners' Group is developing generic guidelines for emergency operating procedures regarding extended loss of feedwater flow as part of its response to the requirements specified in NUREG-0578 concerning inadequate core cooling. This effort includes supporting analyses which are being used to obtain more definitive information about plant response and the time at which feedwater flow must be reestablished in order to prevent core damage. Our evaluation regarding inadequate core cooling will be published in a separate report.

6. SUMMARY OF FEEDWATER-RELATED EVENTS

We reviewed occurrences since 1976 involving feedwater malfunctions at each of the operating W-designed plants. When an incident violates plant Technical Specifications, it is reported in a Licensee Event Report. Generally, events occurring during startup, and other minor equipment failures, are not reportable.

Each licensee with an NSSS designed by W provided information at our request pertaining to all events that resulted in a complete loss of main feedwater over the last three years of operation. These events are summarized in Tables II-2 and II-3. Events which occurred during reactor startup and those that did not result in a feedwater transient were not reported. It should be noted that this list is incomplete, since some other initiating events which resulted in a loss of main feedwater may not have been included. Therefore, the events identified were used to obtain a minimum frequency of loss of feedwater events.

6.1 Beaver Valley Unit 1

Beaver Valley has had four loss of feedwater events. They are described below. During the events, safety systems were available to perform their functions as required. No significant consequences have been reported.

Unit No.	Date	% Power - Cause of the Event
1	7/30/76	At 30% power, the main feedwater pump tripped due to low oil pressure caused by a leak in the oil supply to the bearings. The leak was caused by high vibration in the feedwater pumps due to inadequate recirculation flow and impeller design. The reactor tripped on a low-low steam generator water level.
1	8/19/76	At 17% power, the main feedwater pumps tripped on low suction pressure. Previously at 30% power, the feedwater pump dropped in suction pressure, and the load was reduced. The "A" condensate pump was shut down; this further dropped the feedwater pump suction pressure. The condensate pump was restarted, but it tripped on overcurrent and the main feedwater pumps tripped on low suction pressure. The cause of the event was clogged suction strainers on the condensate pumps. The reactor tripped on a low-low steam generator water level.
1	7/2/78	At 55% power, the strainers on the "B" condensate pump had been cleaned and the pump had been placed back in service. The "A" condensate pump was shut down and isolated. The feedwater pumps tripped on low suction pressure caused by a loss of a set screw on the valve position indicator on the condensate suction valve. The valve position indicator was indicating full open, but the valve was actually 60% open, causing the low flow. The reactor was tripped on a low steam generator water level coincider with a steam flow-feed flow mismatch.

Unit No.	Date	% Power - Cause of the Event
1	7/28/78	At 100% power, an electrical fault occurred in the main output transformer. Due to incorrect relay operations, the transformer fault resulted in a loss of all power to the station and a loss of all feedwater pumps. Off-site power was restored 21 minutes later. One emergency diesel generator failed to start due to a faulty field flush relay. The PORV was actuated during the event and performed as designed.

6.2 D. C. Cook 1 & 2

D. C. Cook Unit 1 has had six loss of feedwater events and Unit 2 has had a total of seven events. The licensee has listed events that occurred above 10% power levels and only when the event was directly linked to the main feedwater pump trip. The initiating cause of the loss of the feedwater pump was not provided.

Unit No.	Date	% Power - Cause of the Event
1	2/8/76	At 22% power, cause undefined. The reactor was tripped on a low steam generator water level with steam flow-feed flow mismatch.
1	3/26/77	At 98% power, same as 2/8/76 event.
1	2/11/78	At 18% power, same as 2/8/76 event.
1	3/7/78	At 100% power, same as 2/8/76 event.
1	12/12/78	At 100% power, same as 2/8/76 event.
2	3/28/78	At 29% power, cause undefined. Reactor tripped on low-low steam generator water level.
2	4/6/78	At 30% power, same as 3/28/78 event.
2	4/9/78	At 41% power, same as 3/28/78 event.
2	6/14/78	At 65% power, cause undefined. Reactor tripped on low steam generator water level coincident with a steam flow-feed flow mismatch.
2	7/14/78	At 90% power, same as 6/14/78 event.
2	11/26/78	At 88% power, same as 6/14/78 event.

Unit No.	Date	% Power - Cause of the Event
2	1/3/79	At 100% power, cause undefined. Manual trip actuated because main feed flow had been valved off to both steam generators. Steam generator water level had not yet reached its automatic trip setpoint.
1	4/6/79	At 96% power. Loss of both main feedwater pumps on low main condenser vacuum.

6.3 H. B. Robinson

H. B. Robinson has not reported any loss of main feedwater events in its last three years of operation.

6.4 Haddam Neck

Haddam Neck has not reported any loss of main feedwater events in its last three years of operation.

6.5 Indian Point 2 and 3

Indian Point Unit 2 has had two loss-of-feedwater events, and Unit 3 has had a total of 11 events. They are described below. During each event, the safety systems were available to perform their safety function as required. No significant safety consequences resulted. Indian Point Unit 3 has had other feedwater interruptions that have resulted in the loss of feedwater, but they have not been included in the 11 events listed here. Ten high-steam generator level trips and three low-low steam generator water level trips have occurred-mostly at low power levels with the feedwater level controls in manual. These events occurred during the plant's first six months of operation. It appears that the frequency of occurrence has decreased with reactor operational experience.

Unit No.	Date	% Power - Cause of the Event
2	6/14/77	At 100% power, while the lube oil pumps for the main feedwater pumps were being tested, an oil pump did not function properly. The loss of oil pressure caused both feedwater pumps to trip. The reactor tripped on a low steam generator level coincident with a steam flow-feed flow mismatch.
2	2/16/79	At 70% power, both main feedwater pumps tripped from the loss of the lube oil pump. The reactor tripped on a low steam generator level coincident with a steam flow-feed flow mismatch.
3	7/8/76	At 60% power, during power escalation, high level, then low level alarms were received in rapid succession from the heater drain tank, followed by tripping of the heater drain pump. As steam generator levels were decreasing, the flow control ΔP valve was overridden open, which caused the level in the steam

Unit No.	Date	% Power - Cause of the Event
		generator to rise rapidly. The reactor tripped on a high steam generator water level.
3	7/19/76	At 26% power, during a turbine load rejection test (75% power to 25% power), a heater drain pump tripped due to low flow. The reactor tripped on a high steam generator water level.
3	11/17/76	At 91% power, moisture from a packing leak shorted out a solenoid valve, causing the feedwater regulating valve to close. The reactor tripped on a low steam generator water level coincident with a steam flow-feed flow mismatch.
3	12/9/76	At 91% power, same as the 11/17/76 event. Caused by steam condensation from a prior packing gland leak.
3	12/17/76	At 58% power, the high pressure steam valve failed to respond to control speed (valve was in the full shut position with the demand feed for full open) causing the levels in the steam generators to decrease. The reactor was tripped on a steam generator low water level coincident with a steam flow-feed flow mismatch.
3	1/10/77	At 100% power, the heater drain pump tripped on low flow (the heater drain pump regulating flow control valve disk separated from its stem, causing the valve to close), causing feedwater pump suction pressure cutback. The reactor tripped on a low-low steam generator water level.
3	2/20/77	At 91% power, the heater drain pump tripped, resulting in an uncontrolled load pick-up on machine, and then the second heater drain pump tripped. The reactor tripped on steam generator level coincident with a steam flow-feed flow mismatch.
3	2/10/78	At 25% power, while increasing load, the operator increased main feedwater pump speed. The increase in speed caused the pump to trip on overspeed. The reactor tripped on low level in the steam generators.
3	3/28/78	At 91% power, as the main feedwater pumps were in manual control, the feedwater pump flow started to decrease. Manual control was unable to keep up. The reactor tripped on a low-low steam generator water level.
3	4/28/78	At 91% power, a leak in the level column A flashed, resulting in an erroneous high level in the heater drain tank. The signal caused the pump regulators to open and the tank was

Unit No.	Date	% Power - Cause of the Event
		pumped out, tripping both heater drain pumps. The reactor tripped on low-low steam generator water level.
3	12/18/78	At 100% power, erratic operation of both main feedwater pumps caused the pumps to trip. The reactor was tripped on low level in all steam generators. Erratic operation was due to dirt accumulation in controls.

6.6 Farley Unit 1

Farley Unit 1 has had 15 loss of feedwater events. They are described below. All events listed by the licensee caused a feedwater pump trip on low suction pressure and safety systems performed their function as required. No significant safety consequences resulted.

Unit No.	Date	% Power - Cause of the Event
1	8/25/77	At 20% power, a feedwater pump tripped on low suction pressure due to only one condensate pump running during startup. Reactor tripped on a low-low steam generator water level.
1	8/25/77	At 17% power, same as the above event, except the turbine was tripped manually, resulting in a reactor trip.
1	8/30/77	At 20% power, same as the initial 8/25/77 event. Procedures were revised to require two condensate pumps running during startup.
1	9/18/77	At 74% power, low hotwell level cavitated the condensate pumps, causing feedwater pump trip on a low suction pressure. Hotwell level control was in manual. The reactor tripped on a low-low steam generator water level.
1	10/3/77	At 48% power, power was lost to controllers for the miniflow valve, causing the valve to fail open. The flow increase caused pump runout and loss-of-suction pressure to the feedwater pumps. The reactor tripped on a low-low steam generator water level.
1	11/17/77	At 100% power, the condensate recirculation valve failed open due to a loss of power to the valve from a loose fuse clip in the power panel. The reactor tripped on a low steam generator water level coincident with a steam flow-feed flow mismatch.
1	1/19/78	At 100% power, the condenser level controller malfunctioned, causing a feedwater pump trip. The reactor tripped on a low-low steam generator water level.

Unit No.	Date	% Power - Cause of the Event
1	2/9/78	At 100% power, feedwater pumps tripped on low suction pressure due to malfunctioning of feedwater heaters and drain tank level controllers. Reactor tripped on a low-low steam generator water level.
1	2/15/78	At 100% power, inadvertant opening of a breaker caused loss of solenoids in the turbine building. The reactor tripped on a low-low steam generator water level.
1	3/25/78	At 100% power, a feedwater pump tripped due to a faulty coil. The reactor tripped on a low-low steam generator water level.
1	5/26/78	At 100% power, the reactor tripped while a test procedure was being performed. The feedwater pump tripped when a test push button was activated due to a faulty coil.
1	6/6/78	At 95% power, a circulating water pump tripped. The power decrease caused flashing in the heater drain tank and cavitation of the heater drain pumps, which subsequently tripped the feedwater pumps on low suction pressure. The reactor tripped on a low steam generator water level coincident with a steam flow-feed flow mismatch.
1	7/16/78	At 89% power, a circulating water pump tripped due to grounding of all phases. The reactor was manually tripped.
1	9/15/78	At 22% power, the high pressure steam supply to a feedwater pump was isolated due to excessive steam leakage. The reactor tripped on a low steam generator water level coincident with a feed flow-steam flow mismatch.
1	12/31/78	At 100% power, the feedwater pump suction header pressure switch instrument valves were isolated due to a faulty micro switch.

6.7 Kewaunee

Kewaunee has had two loss-of-feedwater events. They are listed below. No safety consequences have resulted.

Unit No.	Date	% Power - Cause of the Event
1	1/22/76	At 100% power, undervoltage on a bus caused the loss of feedwater and the loss of a reactor coolant pump with a subsequent reactor trip.

Unit No.	Date	% Power - Cause of the Event
1	6/6/77	At 100% power, lightning struck the substation. This caused the same results as the 1/22/76 event.

6.8 North Anna Unit 1

North Anna Unit 1 has had 11 loss of feedwater events as listed below. No adverse safety consequences have been reported, and safety systems functioned as required.

Unit No.	Date	% Power - Cause of the Event
1	4/18/78	At 10% power, an error in the nuclear power range current setting was twice what it should have been. This setting is an input to the automatic bypass steam generator level control system. At power, the signal to the level controller was doubled, causing a high water level in the steam generators. This caused a turbine trip and a reactor trip.
1	4/19/78	Power not stated. While blowing down the steam dump lines, debris stirred up from the hotwell and clogged the main condensate pump suction strainers. The feedwater pumps tripped on low suction pressure. The reactor was tripped on a low steam generator water level.
1	5/11/78	Power not stated. With the steam generator water level control in manual, the water level could not be controlled properly with increased steam flow. The reactor was tripped on high steam generator water level.
1	5/11/78	At 99% power, steam flow caused the main steam valves to begin flapping. The main steam flow forced the valves closed when they fell into the path of the flow stream.
1	5/13/78	Power not stated. The valve positioner vibrated off the steam generator feed regulating valve. Feedback arm indicated to the control system that the valve was closed. The system opened the valve fully, causing a high water level in the steam generator. The reactor tripped on the high-high level signal.
1	5/31/78	Power not stated, same as 5/11/78 event.
1	5/31/78	At 99% power, same as 5/11/78 event valve closing.
1	6/6/78	Power not stated, same as 5/13/78 event.

Unit No.	Date	% Power - Cause of the Event
1	7/8/78	At 10% power, debris stirred up in the main condensate hotwell and clogged the condensate suction strainers. The main feedwater pumps tripped on low suction pressure. The reactor tripped on a low steam generator water level.
1	3/30/79	Power not stated. The unit was ramped down due to loss of cooling water from "A" reactor coolant pump causing a manual reactor turbine generator trip due to rapid pressurizer level and pressure drop. Low levels in the steam generators occurred, causing a reactor trip.
1	5/4/79	Power not stated. The main feedwater pump tripped due to loss of oil pressure. The reactor tripped on a low steam generator water level coincident with a feed flow-steam flow mismatch.
1	9/25/79	At 78% power, a tube rupture occurred inside a drain cooler in one of the feedwater heaters. A drain cooler pump valve apparently failed closed, causing a condensate backup. The condensate backup caused a turbine trip on high feedwater heater level. The reactor tripped on the interlock requiring reactor trip when a turbine trip occurs above 15% power. A turbine bypass valve stuck open, causing rapid plant cooldown until the operator shut the main steam isolation valves, stopping steam flow through the stuck-open valve. Rapid depressurization caused initiation of safety injection, leading to system pressure increase and opening of the PORVs. One main feedwater pump was tripped manually at reactor trip, the other feedwater pump was tripped off at the safety injection signal. Auxiliary feedwater system started on low-low steam generator level and functioned as designed.

6.9 Point Beach 1 & 2

Point Beach Units 1 and 2 have not reported any loss of main feedwater events in the last three years of operation.

6.10 Prairie Island 1 and 2

Prairie Island Unit 1 has had one loss of feedwater event and Unit 2 has experienced a total of two. No adverse safety consqequences were reported. Safety systems functioned as required.

Unit No.	Date	% Power - Cause of the Event
1	8/31/78	At 100% power, the main feedwater pump motor windings shorted and an overcurrent tripped the pump. The reactor tripped on low steam generator water level.

Unit No.	Date	% Power - Cause of the Event
2	2/11/76	At 100% power, an instrument and control procedural check caused feedwater isolation valve closure and a subsequent feedwater trip. The reactor tripped on low steam generator water level.
2	11/10/77	At 90% power, the feedwater pumps tripped due to the loss of lube oil. The reactor tripped on low steam generator water level.

6.11 Ginna

Ginna has not reported any loss of main feedwater events in its last three years of operation.

6.12 Salem 1

Salem Unit 1 has had a total of four loss of feedwater events. They are listed below. No adverse safety consequences were reported and safety systems functioned as required.

Unit No.	Date	% Power - Cause of the Event
1	12/27/76	At 18% power, the feedwater pumps tripped on low suction pressure (no reason stated). The reactor tripped on a low-low steam generator water level.
1	12/27/76	At 17% power, same as above.
1	8/13/77	At 50% power, with feedwater pumps in manual control, control problems resulted in a feedwater pump trip. The reactor was tripped on a low-low steam generator water level.
1	12/5/77	At 55% power, the loss of the condensate pumps caused a feedwater trip due to low suction pressure. The reactor was tripped on a low-low steam generator water level.

6.13 San Onofre Unit 1

San Onofre Unit 1 has not reported any loss of main feedwater events in the last three years of operation.

6.14 Surry Units 1 and 2

Surry Units 1 and 2 have each had two loss of feedwater events. They are listed below. In all the events, the safety systems functioned as required. No significant safety consequences were reported.

Unit No.	Date	% Power - Cause of the Event
1	3/10/76	At 100% power, the operating high pressure drain pump was inadvertantly lost. A simultaneous automatic restart of both high pressure drain pumps caused a high steam generator water level. The reactor tripped and feedwater isolation occurred from a steam line to steam head ΔP safety injection.
1	12/9/78	At 12% power, the main feed regulating valve caused excessive feeding to its steam generator. The reactor tripped on a high-high steam generator water level.
2	9/15/76	At 100% power, a manual safety injection actuation signal was performed during a steam generator tube leak.
2	4/23/77	At 100% power, same as the 3/10/76 event.

6.15 Trojan

Trojan has had three loss of feedwater events. They are as listed below. The AFW system was automatically initiated during each event and performed its intended safety function. No adverse consequences have been reported.

Unit No.	Date	% Power - Cause of the Event
1	11/27/76	At 100% power, a feedwater pump tripped due to overspeed. The reactor was tripped on a low steam generator water level coincident with a steam flow-feed flow mismatch.
1	12/6/76	At 35% power, (reduction from 100% to 35% power was caused by pump overspeed trip) oscillations of the heater drain pump flow control valve caused a feedwater pump trip on low suction pressure. The reactor was tripped on a low steam generator water level coincident with a steam flow-feed flow mismatch.
1	6/21/77	At 60% power, the feedwater pump tripped on low suction pressure (cause undefined). The reactor tripped on a low-low steam generator water level.

6.16 Turkey Point 3 and 4

Turkey Point Unit 3 has had only one loss of feedwater event and Unit 4 has had a total of four, as listed below. There were no adverse safety consequences resulting from the events. The protection systems and the alternate systems functioned as designed.

Unit No.	Date	% Power - Cause of the Event
3	7/23/77	At 29% power, a disconnect switch on the Davis Line arced to ground, causing a turbine trip. The reactor was manually tripped after the load rejection. The startup transformer breakers were locked out on the automatic transfer due to transfer times after generator breaker opening. The unit underwent natural circulation cooling for about 21 minutes.
4	7/26/76	At 100% power, the loss of a feedwater pump caused a turbine runback. The sudden load rejection caused the steam generator levels to shrink. The reactor was tripped on a low-low steam generator water level. Normally, after a turbine/reactor trip, the power supply is transferred from the auxiliary transformer to the startup transformer. Due to a faulty switch in the startup breaker, this transfer was incomplete. Reactor coolant pumps were unavailable. The unit was cooled by natural circulation until the reactor coolant pumps and feedwater pumps were restarted.
4	12/4/76	At 100% power, an inverter failed, causing the loss of voltage on a vital instrumentation ac bus. The loss of voltage placed several protection and safeguard circuits in the tripped mode, transferred steam generator auto/manual controls to manual with their output signal fixed "as-is," and caused a spurious dropped rod signal. This initiated a 30% turbine runback, which produced a reactor trip on high-high steam generator water level. The turbine was also tripped.
4	8/7/77	At 75% power, the breaker located in the vital instrumentation ac bus tripped. When the voltage was lost, the feedwater flow signal failed low. The feedwater regulator valve went to full open, causing a rapid increase in steam generator level. The reactor tripped on high-high steam generator water level. The turbine also tripped.
4	7/7/78	At 100% power, the feedwater regulator valve failed in the full open position, due to moisture from a packing leak. The reactor was tripped on a high-high steam generator water level.

6.17 Yankee Rowe

Yankee Rowe has not reported any loss of main feedwater events in its last three years of operation.

6.18 Zion 1 and 2

Zion Unit 1 has had a total of two loss of feedwater events and Unit 2 has had one loss of feedwater event as listed below. There were no adverse consequences resulting from the events. The reactor protection systems functioned as designed and the alternate systems functioned normally.

Unit No.	Date	% Power - Cause of the Event
1	5/13/78	At 88% power, a leaking roof drain caused water to spill on a motor control center that provided power to the oil supply pump motor to the turbine-driven feedwater pumps. The water spill caused both oil supply pumps to trip. The feedwater pumps were tripped on low bearing oil trip. This resulted in a turbine trip and a reactor trip.
1	3/5/79	At 95% power, the "B" turbine-driven feedwater pump dropped in speed due to a loss of control oil pressure. The reactor was ramped down to 38% power. The supply valve that provides high pressure steam to the "C" turbine driven pump failed closed. The "C" turbine driven pump dropped in speed, causing a turbine trip and a reactor trip.
2	12/4/78	At 9% power, the turbine-driven feedwater pump failed to transfer to the high pressure steam supply. This caused a turbine trip and a reactor trip due to the loss of feedwater.

TABLE II-3

SUMMARY OF LOSS OF FEEDWATER EVENTS SINCE 1976 BY SYSTEM FOR W-DESIGNED OPERATING PLANTS

System	Number	Total for each system
Feedwater Pump System		43
(a) cause undefined	16	
(b) cause unknown		
(c) feedwater controls	12	
(d) lube oil subsystem	6	
(e) pump overspeed	2	
(f) loss of pump drive		
(1) electric supply	4	
(2) steam supply	3	
Condensate System		10
(a) demineralizers	0	
(b) precoat filters	0	
(c) pump and strainers	7	
(d) valves	3	
Heaters and System		8
(a) control valves	3	
(b) drain tank and pumps	5	
Electrical System		9
(a) faults	4	
(b) loss of power	1	
(c) bus failure	3	
(d) lightning strike	1	
Circulating Water System		4
(a) pumps	2	
(b) level control	2	
Main Steam System		3
Turbine System		1
Reactor Coolant System		1
Errors		1
	Total	80

TABLE II-4

LOSS OF MAIN FEEDWATER EVENTS SINCE 1976
FOR W-DESIGNED PLANTS BY PLANT*

Plant Name	Reactor Years of Operation 1976	1977	1978 to Date	Total Number of Events Since 1976
Beaver Valley 1	2	0	2	4
D.C. Cook 1	1	1	4	6
D.C. Cook 2	-	6	1	7
H. B. Robinson	0	0	0	0
Haddam Neck	0	0	0	0
Indian Point 2	0	1	1	2
Indian Point 3	5	2	4	11
Farley	-	6	9	15
Kewaunee	1	1	0	2
North Anna	-	-	11	11
Point Beach 1	0	0	0	0
Point Beach 2	0	0	0	0
Prairie Island 1	0	0	1	1
Prairie Island 2	1	1	0	2
Ginna	0	0	0	0
Salem	2	2	0	4
San Onofre	0	0	0	0
Surry 1	1	0	1	2
Surry 2	1	1	0	2
Trojan	2	1	0	3
Turkey Point 3	0	1	0	1
Turkey Point 4	2	1	1	4
Yankee Rowe	0	0	0	0
Zion 1	0	0	2	2
Zion 2	0	0	1	1
	18	24	38	80

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*W-designed plants have experienced 80 loss-of-feedwater events in the last three years and five months of operation. The 25 plants have 78 years of operating history.

APPENDIX III

AUXILIARY FEEDWATER SYSTEMS

1. INTRODUCTION

The Three Mile Island Unit 2 (TMI-2) accident and subsequent investigations and studies highlighted the importance of the auxiliary feedwater (AFW) system in the mitigation of transients and accidents. As part of its assessment of the TMI-2 accident and related implications for operating plants, the staff evaluated the AFW systems for all operating plants having nuclear steam supply systems (NSSS) designed by Westinghouse (W) (25 units) or Combustion Engineering (CE) (8 units). (See note below.)

The objectives of this study were to: (1) identify necessary changes in AFW system design or related procedures at these plants in order to assure the continued safe operation of these plants, and (2) to identify other system characteristics in design of the AFW system for these plants which, on a long term basis, may require system modifications. To accomplish these objectives, we:

(1) Reviewed plant-specific AFW system designs in light of current regulatory requirements, and

(2) Assessed the relative reliability of the various AFW systems under various loss of feedwater transients (one of which was the initiating event at TMI-2) and other postulated potential failure conditions by determining the potential for AFW system failure due to common causes, single point vulnerabilities and human error.

As part of our evaluation, we performed a standard deterministic type of safety review, using as principal guidance the acceptance criteria specified in Section 10.4.9 of the Standard Review Plan (SRP) (1). In conjunction with this deterministic review, we used event tree and fault tree logic techniques, as part of a reliability analysis to determine dominant failure modes and assess AFW system comparative reliability levels under specified types of transients. When the recommendations identified in this review are implemented, the reliability of the AFW systems for each operating plant should be improved, with the degree of improvement dependent upon whether the AFW systems were initially characterized as having relatively high or low reliabilities (see Section 4.6 of this appendix for details).

The time and personnel limitations imposed on this study precluded a complete and extensive review of each AFW system. The review was based primarily upon information provided by each licensee at a four-hour meeting with the staff review team (composed of a systems engineer and a reliability engineer) to review the as-built AFW system design and operation. Consequently, the results should be viewed in terms of the general conclusions and insights, and not as an absolute reliability analysis of generic or plant-specific AFW systems upon

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NOTE: Studies of the AFW systems at Babcock and Wilcox (B&W) designed-operating plants were subjects of separate Commission orders and other work performed by the NRC staff.

which the acceptability of these AFW system designs may be judged. This reliability assessment resulted in the development of generic and plant-specific recommendations to improve AFW system reliability. It was recognized that it would be very difficult and subject to large uncertainty if an attempt was made to quantify the reliability improvement inherent through implementation of the recommended actions. It was decided that such an effort was outside the scope of this study.

Some AFW systems in operating nuclear power plants do not meet all current staff licensing criteria contained in the Standard Review Plan. The degree of conformance varies with the age and specific plant design of the 33 units addressed in this study. For example, 10 architect/engineering organizations were involved in the plant design and construction of these 33 units. A specific objective of this study was to determine whether the lack of conformance with any of these later requirements represented potential safety problems, considering the TMI-2 experience. The recommendations identified in this study reflect areas of potential weaknesses where changes to improve AFW system reliability should be implemented.

The results of the AFW system design review and the evaluation of TMI-2 accident implications were judged to require consideration for corrective action if any one of the following conditions was identified:

(1) Common mode failures (particularly those related to human error),

(2) Single point failures, or

(3) Any dominant causes of AFW system unreliability.

Our limited review focused on the implications of the TMI-2 accident, particularly human errors, and thus we did not reevaluate the design basis for each AFW system, nor did we focus upon all possible system interactions that could affect AFW system reliability. However, if the information suggested a potential for loss of AFW from such causes, this potential was noted during the specific plant reviews, with followup evaluations recommended to determine the need for additional actions.

In determining which safety issues required short-term licensing action versus those that could be deferred for further evaluation, we used simplified engineering evaluations and qualitative judgment of the safety significance of the various issues. In this regard, we recommended actions if their implementation would provide substantial, additional protection required for the public health and safety. The recommended actions were specific and safety-significant in their character, could be implemented in a timely manner, and would not likely be overturned or contradicted by continuing studies or investigations. Some of them may eventually be displaced, however, by more comprehensive long-term changes in nuclear power plant regulation. In some cases, based on information or analysis developed to date, it is not clear that a basis for a decision is available. In such cases, we have judged the item to be of sufficient safety significance to require an early commitment to get studies or testing under way to provide a basis for resolution of the issue. As required, the recommended action is to obtain a commitment for a longer-term modification, study, or test by affected licensees.

2. PURPOSE AND DESCRIPTION OF AUXILIARY FEEDWATER SYSTEM

The AFW system functions as an emergency system for the removal of heat from the primary system when the main feedwater system is not available. It also plays an important role in mitigating the effects of some design basis events; for example, some small break loss-of-coolant accidents (LOCAs). The AFW system is designed to hold the plant at hot standby, or to cool down the primary system to temperature and pressure levels at which the low pressure decay heat removal system can operate. The AFW system can also be used during normal plant startup and shutdown conditions. AFW systems usually consist of a combination of steam turbine-driven and electric motor-driven pumps. The AFW system can provide, with any one pump out of service, enough water to the steam generators for decay heat removal following loss of main feedwater flow. Table III-1 provides a summary of the pump combinations, flow ratings and modes of initiation for the AFW system for each plant reviewed. Appendix X provides specific AFW system descriptions, a simplified flow sheet for each W operating plant, and an evaluation with corresponding recommendations.

3. DETERMINISTIC EVALUATION

3.1 Background and Objectives

In our review of current applications for construction permits and operating licenses for pressurized water reactors, we evaluate the AFW system to assure that the design conforms to the applicable General Design Criteria (GDC) in Appendix A to 10 CFR Part 50. The General Design Criteria identified in Section 10.4.9 of the Standard Review Plan applicable to the AFW system design are listed below:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as related to structures housing the system, and the system itself being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods.

(1) GDC 4, "Environmental and Missile Design Bases," with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip and jet impingement forces associated with pipe breaks.

(3) GDC 5, "Sharing of Structures, Systems, and Components," as related to the capability of shared systems and components important to safety to perform required safety functions.

(4) GDC 19, "Control Room," as related to the design capability of system instrumentation and controls for prompt hot shutdown of the reactor, and potential capability for subsequent cold shutdown.

TABLE III-1. AUXILIARY FEEDWATER SYSTEMS AT WESTINGHOUSE-DESIGNED OPERATING PLANTS AND COMBUSTION ENGINEERING - DESIGNED OPERATING PLANTS

Westinghouse-Designed Plants

Plant	No. of Pumps/ Type of Drive	Capacity	AFWS Mode of Initiation	Comments
Beaver Valley 1	1-Steam Driven	Steam: 700 gpm @ 2696 ft.	Automatic	
	2-Motor Driven	Motor: (each) 350 gpm @ 2696 ft		
D. C. Cook 1 & 2	1-Steam Driven*	Steam: 900 gpm @ 2714 ft	Automatic	Per unit motor pumps supply both units
	1-Motor Driven*	Motor: 450 gpm @ 2714 ft		
Farley 1	1-Steam Driven	Steam: 700 gpm @ 1268 psig	Automatic	
	2-Motor Driven	Motor: (each) 350 gpm @ 1268 psig		
Ginna	1-Steam Driven	Steam: 400 gpm @ 1131 psig	Automatic	
	2-Motor Driven (normal AFWS)	Motor: (each) 200 gpm @ 1114 psig		
	2-Motor Driven (standby AFWS	Motor: 200 gpm	Manual	
Haddam Neck	2-Steam Driven	Steam: 450 gpm @ 1000 psia	Manual	
H. B. Robinson	1-Steam Driven	Steam: 600 gpm @ 1300 psi	Automatic	
	2-Motor Driven	Motor: (each) 300 gpm @ 1300 psi		
Indian Pt. 2 & 3	1-Steam Driven*	Steam: 800 gpm @ 1350 psig	Automatic	per unit
	2-Motor Driven*	Motor: (each) 400 gpm @ 1350 psig		
Kewaunee	1-Steam Driven	Steam: 240 gpm @ 2850 ft	Automatic	
	2-Motor Driven	Motor: (each) 240 gpm @ 2850 ft		
North Anna 1	1-Steam Driven	Steam: 700 gpm @ 2800 ft	Automatic	
	2-Motor Driven	Motor: (each) 350 gpm @ 2800 ft		
Prairie Island 1 & 2	1-Steam Driven*	Steam: 220 gpm @ 1200 psig	Automatic	Per unit motor pump normally feeds opposite unit steam generators
	1-Motor Driven*	Motor: 220 gpm @ 1200 psig		
Pt. Beach 1 & 2	1-Steam Driven*	Steam: 400 gpm @ 1192 psig	Automatic	Per unit motor pump supplies both units
	1-Motor Driven*	Motor: (each) 200 gpm @ 1192 psig		

*Note: See Comments column

TABLE III-1 (Continued)

Westinghouse-Designed Plants

Plant	No. of Pumps/ Type of Drive	Capacity	AFWS Mode of Initiation	Comments
Salem 1	1-Steam Driven	Steam: 880 gpm @ 1550 psi	Automatic	
	2-Motor Driven	Motor: (each) 440 gpm @ 1300 psi		
San Onofre 1	1-Steam Driven	Steam: 300 gpm @ 1110 psi	Manual	
	1-Motor Driven	Motor: 235 gpm @ 1035 psi		
Surry 1 & 2	1-Steam Driven*	Steam: 700 gpm @ 2730 ft	Automatic	One pump each AFW system can feed opposite unit
	2-Motor Driven*	Motor: (each) 350 gpm @ 2730 ft		
Trojan	1-Steam Driven	Steam: 960 gpm @ 3400 ft	Automatic	
	1-Diesel Driven	Diesel: 960 gpm @ 3400 ft		
Turkey Pt. 3 & 4	3-Steam Driven* for both units	(each) 600 gpm @ 2775 ft	Automatic	One pump normally supplies each unit - 3rd pump is backup for either unit
Yankee Rowe	1-Steam Driven*	Steam: 90 gpm @ 1200 psi	Manual	*charging and safety injection systems serve as backup
Zion 1 & 2	1-Steam Driven*	Steam: 900 gpm @ 3099 ft	Automatic	per unit
	2-Motor Driven*	Motor: (each) 450 gpm @ 3099 ft		

*Note: See Comments column

TABLE III-1 (continued)
AUXILIARY FEEDWATER SYSTEMS

Combustion Engineering-Designed Plants

Plant	No. of pumps/ Type of Drive	Capacity	AFWS Mode of Initiation
ANO	1-Steam Driven 1-Motor Driven	Steam: 575 gpm @ 2800 ft. Motor: 575 gpm @ 2800 ft.	Automatic
Calvert Cliffs 1&2	2-Steam Driven per unit	700 gpm @ 1100 psia each	Manual
Ft. Calhoun 1	1-Steam Driven 1-Motor Driven	Steam: 260 gpm @ 2400 ft. Motor: 260 gpm @ 2400 ft.	Semi-automatic motor-driven pump manually connected to diesel generator
Main Yankee	1-Steam Driven 2-Motor Driven	Steam: 500 gpm @ 1100 psig Motor: (each) 1500 gpm @ 1100 psig	Manual
Millstone 2	1-Steam Driven 2-Motor Driven	Steam: 600 gpm @ 2437 ft. Motor: (each) 300 gpm @ 2437 ft.	Manual
Palisades	1-Steam Driven 1-Motor Driven	Steam: 415 gpm @ 2730 Motor: 415 gpm @ 2730 ft.	Manual
St. Lucie 1	1-Steam Driven 2-Motor Driven	Steam: 500 gpm @ 1200 psi Motor: (each) 250 gpm @ 1200 psi	

(5) GDC 44, "Cooling Water," to assure the capability to transfer heat loads from the reactor system to a heat sink under all operating conditions, redundancy of components so that the safety function can be performed assuming a single active component failure, and the capability to isolate components or piping, if required, so that the system safety function will be maintained.

(6) GDC 45, "Inspection of Cooling Water Systems," as related to design provisions made to permit periodic inservice inspection of system components and equipment.

(7) GDC 46, "Testing of Cooling Water System," as related to design provisions made to permit appropriate functional testing of the system and components to assure operability and performance of components, and capability of the integrated system to function as intended during all operating conditions.

In determining whether the AFW system designs for such applications meet these General Design Criteria, the staff uses Section 10.4.9 of the Standard Review Plan and Branch Technical Position ASB 10-1 (hereafter referred to as ASB 10-1) as guidance. These documents contain the acceptance criteria for the AFW system and the review procedures to be used by the staff to determine if these acceptance criteria are met. If the staff concludes that the acceptance criteria are met, then it is also able to conclude that the requirements of the applicable General Design Criteria are satisfied.

It was recognized at the outset of this assessment of operating plants that many of these plants do not meet each of the acceptance criteria of the Standard Review Plan, including Section 10.4.9 and ASB 10-1, which relate to the AFW system design. This situation exists because the operating licenses for many of these plants were issued prior to the publication of the Standard Review Plan and, for some of these plants, prior to the publication of the General Design Criteria in February 1971.* The Standard Review Plan was originally issued in November 1975 and revised in 1978.

When the staff issues new or revised regulatory requirements and guidance, it addresses whether the new or revised requirements or guidance should be backfitted to operating plants, as well as plants undergoing licensing review. This decision is guided by Section 50.109 of 10 CFR Part 50 of the Commission's regulations. This regulation states that, if a finding is made to the effect that new requirements provide substantial, additional protection which is required for public health and safety, they are to be backfitted on plants with operating licenses (operating plants). Such a finding has not been made for several requirements contained in SRP Section 10.4.9, which applies to AFW systems. Consequently, as noted above, the AFW systems at some operating plants do not meet all of the requirements imposed on later designs.

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*Although the GDC were promulgated as part of 10 CFR Part 50 in 1971, the basic safety considerations embodied in the GDC had been in general use from the early 1960's.

It should be noted that AFW system designs which do not meet the criteria in the Standard Review Plan are not necessarily in conflict with the General Design Criteria. Deviations from the Standard Review Plan may be justified (even on new plants) provided that an acceptable level of protection is provided in the overall plant design. Prior to our assessment, specific documentation of deviations from the Standard Review Plan, Section 10.4.9, and ASB 10-1 had not been required for all operating plants.

It is against this background that we developed guidelines for the deterministic review of AFW system designs for operating plants. These guidelines are provided below:

(1) Determine the extent to which the AFW system designs meet the criteria of the current Standard Review Plan.

(2) Where AFW system designs do not meet the Standard Review Plan criteria, determine whether changes can be identified that will significantly upgrade the auxiliary feedwater system in operating plants to make them less susceptible to single point failures, human errors, and common mode failures.

(3) Recommend areas of the AFW system design to be evaluated for longer-term improvements in the reliability of AFW system designs.

4. RELIABILITY EVALUATION

4.1 Background and Objectives

The General Design Criteria (GDC) contained in Appendix A to 10 CFR Part 50 provide minimum requirements to be satisfied in the design of nuclear power plants. As such, the GDC provide the basis for the staff's deterministic review of the design features of nuclear power plants, including those of the auxiliary feedwater (AFW) systems. Towards this end, the Standard Review Plan provides criteria and supplemental guidance to the staff for assuring conformance with the GDC, including those applicable to AFW systems. However, to provide additional insight regarding the potential for failures of the AFW system not covered by the Standard Review Plan, the reliability assessments discussed below were performed.

The TMI-2 accident demonstrated that human errors of commission or omission can lead to failures of redundant and diverse AFW system equipment to perform as designed. Thus, the TMI-2 experience tends to confirm past studies(2) indicating human errors are dominant factors(3) in reactor accidents.

Currently, a variety of AFW system designs are being used in the 33 operating plants using W- and CE-designed reactors. This factor gives rise to a variety of hardware dependencies and possible vulnerabilities brought about by human interaction with the design, or possibly some other common influences that could affect AFW system operation. Past studies(2,3) have provided useful engineering insights into those areas of system design where human interactions could significantly affect the availability of standby safety systems. The aforementioned past studies have also provided additional insights for the more probable transient events that tend to dominate the demand for successful operation of the AFW systems.

The reliability assessment approach used and the principal insights and results are summarized below. The comparative reliabilities of the AFW system for the 33 W- and CE-designed operating reactors were evaluated for three different initiation events and are shown in Figure III-4. Figure III-5 presents the comparative reliabilities of the AFW systems for each of the 25 W-designed operating reactors. The results shown in Figures III-4 and III-5 indicate that the reliabilities of the existing AFW system design vary by at least an order of magnitude. The dominant contributors to this variability in reliability were, in general, human errors and single point vulnerabilities as described later. Plant-specific details on these AFW system designs are provided in Appendix X.

4.2 Reliability Assessment Approach and Scope

Reliability techniques and insights were used in this assessment to supplement the more traditional deterministic type of safety review. The principal techniques used in this assessment included the event-tree and fault-tree logic techniques. These techniques are considered(4,5) to represent an acceptable approach in establishing the priorities for the resolution of generic safety issues. These techniques, and the insights derived by use of such logic, have been employed recently to perform a risk-based categorizing and screening of the various generic safety issues.(6)

Accordingly, the staff used the aforementioned techniques to focus on those potential failures that could dominate the unreliability of AFW systems during the following transients.

4.2.1 Loss of Main Feedwater

This transient involves the interruption of the main feedwater flow and the subsequent tripping of the reactor. Reactor experience suggests that about three interruptions of the main feedwater system may be experienced from a number of causes each reactor year.*

4.2.2 Loss of Main Feedwater Due to and Loss of Offsite Power

This transient is initiated by the loss of offsite power which, in turn, causes the interruption of the main feedwater system and the tripping of the reactor. Reactor experience suggests that the main feedwater system may be interrupted by this transient approximately 0.2 to 0.3 times per reactor year.*

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*This number may appear to conflict with the information presented in Table II-4 in Appendix II. As was noted in Section 6 of that appendix, the events listed in Table II-4 represent a minimum frequency of loss of feedwater events, since other initiating events which resulted in a loss of feedwater may not have been included.

4.2.3 Loss of Main Feedwater and Loss of All Alternating Current Power (Station ac Blackout)

This event is initiated by the loss of offsite power, as is the previous transient discussed in 4.2.2 above, except that the onsite emergency alternating current (ac) power sources are also lost. Thus, this scenario represents a significantly degraded case compared to that described in 4.2.2 above. However, since this event requires the concurrent loss of all onsite ac power sources (e.g., usually two diesel-engine generators), its likelihood of occurrence should be orders of magnitude less. This low probability of occurrence notwithstanding, reactor experiences have revealed some precursors to this scenario. For example, in a few instances, all ac power has been lost for periods less than five minutes, thereby encroaching on the time to boil the steam generators dry. In another instance, only one of the redundant onsite emergency ac power sources was available for a period of about 50 minutes. Thus, if for some reason the operating ac source were to fail during this 50-minute interval, this scenario could have occurred.

The ability to cope with this event was not a licensing requirement for the earlier licensed plants. However, the more recently licensed plants, as well as those currently undergoing construction permit or operating license reviews, have been required to provide AFW system designs capable of functioning upon the loss of all ac power. Additionally, the decision whether or not this transient should be a plant design basis, and for what period of time this condition would be assumed to exist, is being reviewed as an unresolved safety issue under Generic Task Action Plan A-44 (7). Because of the above considerations, the ability of each operating plant's AFW system to cope with this transient was included in this assessment.

4.3 Generic Event Trees

The inductive logic used in evaluating the relative reliabilities of the various AFW systems involved the use of generic event trees. The dominant failures affecting the availability of the various AFW system designs for each of the three transients were assessed on a conditional basis rather than on an overall probability basis; i.e., the reliability of each AFW system was calculated, given that the applicable transients described above had occurred. Figure III-1 illustrates an event tree applicable to many of the current AFW system designs. Although this event tree does not contain all the various systems that may become involved over the course of the transient, it illustrates possible accident sequences and outcomes that could result. The heavily shaded sequence illustrates one of the loss of main feedwater transients described above that demands successful operation of the AFW system. The time interval of interest for all the transient events considered is the unavailability of AFW systems during the period of time to boil the steam generators dry. Beyond this interval, primary coolant would be discharged via pressurizer relief and/or safety valves and thereby be lost from the primary coolant system. Without the satisfactory operation of primary coolant makeup systems (e.g., high pressure injection systems), the reactor core could be uncovered and eventually damaged. Further, as this boil-dry time is approached, the ability to drive the steam turbine-driven pumps AFW could be lost. If the AFW system

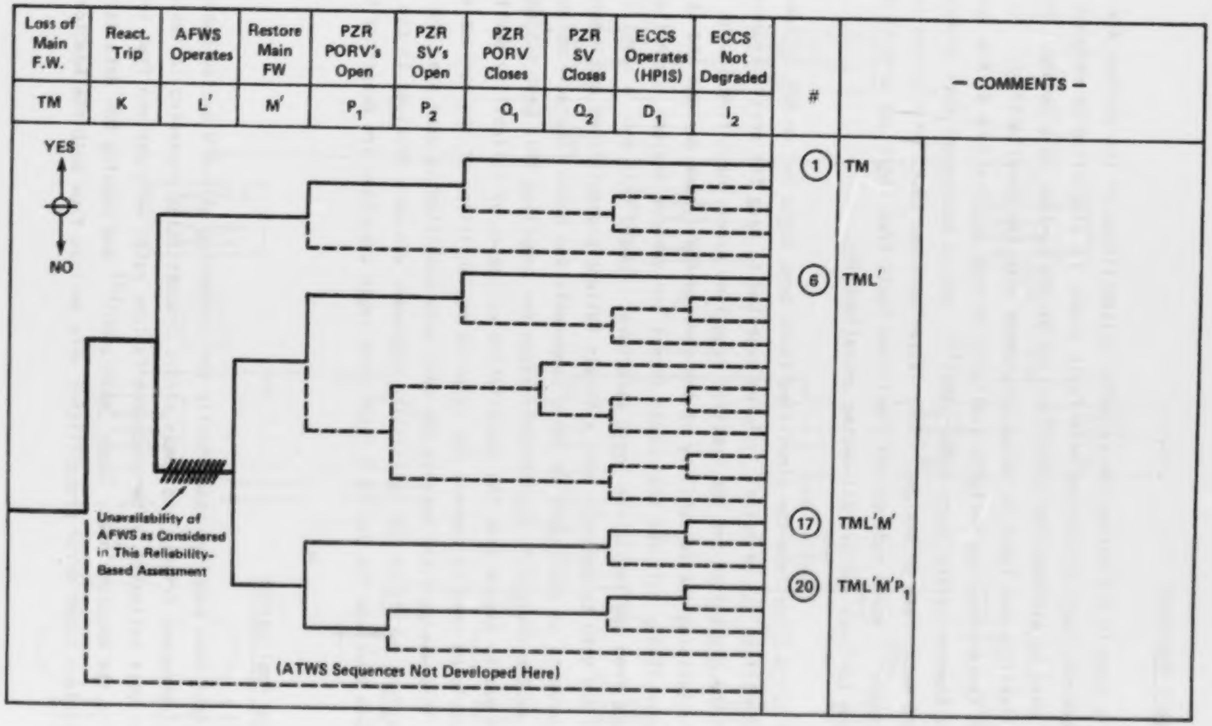
Loss of Main F.W.
React. Trip
AFWS Operates
Restore Main FW
PZR PORV's Open
PZR SV's Open
PZR PORV Closes
PZR SV Closes
ECCS Operates (HPIS)
ECCS Not Degraded
TM
K
L'
M'
P1
P2
Q1
Q2
D1
I2
#
— COMMENTS —
YES
NO
Unavailability of AFWS as Considered in This Reliability-Based Assessment
(ATWS Sequences Not Developed Here)
① TM
⑥ TML'
⑰ TML'M'
⑳ TML'M'P1

Figure III-1 Example Event Tree: Transient Involving Loss of Main Feedwater – Offsite AC Power Available. (Case of LMFW)

design contains only steam turbine-driven pumps, or if the transient is such that only this AFW subsystem is available, and if the boil-dry time is approached, then the likelihood of initiating AFW system operation would be reduced significantly.

4.4 Fault Tree Logic Approach

The deductive logic used in evaluating the relative reliabilities of the various AFW systems was based on the Boolean logic associated with fault trees. A simplified or reduced fault-tree approach was used to estimate the unavailability of AFW systems to a demand. In this assessment, unavailability was taken as being synonomous with the unreliability. This approach relied on the engineering insights available through applications of the system fault trees in the Reactor Safety Study (WASH-1400)(2), and on subsequent work undertaken on additional light water reactor designs. This latter work was based on a system survey and analysis technique(8) where reduced and simplified fault tree logic was used to estimate the dominant system failures and overall system unavailability.

Figures III-2 and III-3 illustrate the simplified fault tree logic for an AFW system design. Each fault tree identifies the principal failures expected to have the most influence on the unavailability of the AFW system for the specific transient event identified in the figure. To assist in characterizing the reliability of the various AFW system designs and to help in identifying the more likely failures that could affect the various designs, quantitative estimates were made from the fault tree logic structure. Toward this end, a specific data base was compiled and used to generate best estimate failure probabilities and human error potentials considered to be applicable to those components and human interactions over the range of the AFW system designs.* A principal reason for compiling this best estimate type of data base was to assure that the quantitative estimates of reliability derived from the fault trees could be used to compare the relative reliabilities of the various AFW system designs. This was possible because the data were consistently used over the range of the AFW system designs by all of the reliability engineers who were involved in the fault analyses. The data base and its use in a fault tree logic structure are described briefly below.

4.5 Data Base and Application

Table III-2 presents data compiled specifically for conducting this AFW system generic assessment. The component failure and human errors probabilities presented in Table III-2 represent current best estimates. The component failure rates were derived from several sources, including the Reactor Safety Study (WASH-1400)(2) and ongoing NRC data assessment programs. The various human error probabilities were derived from both the Reactor Safety

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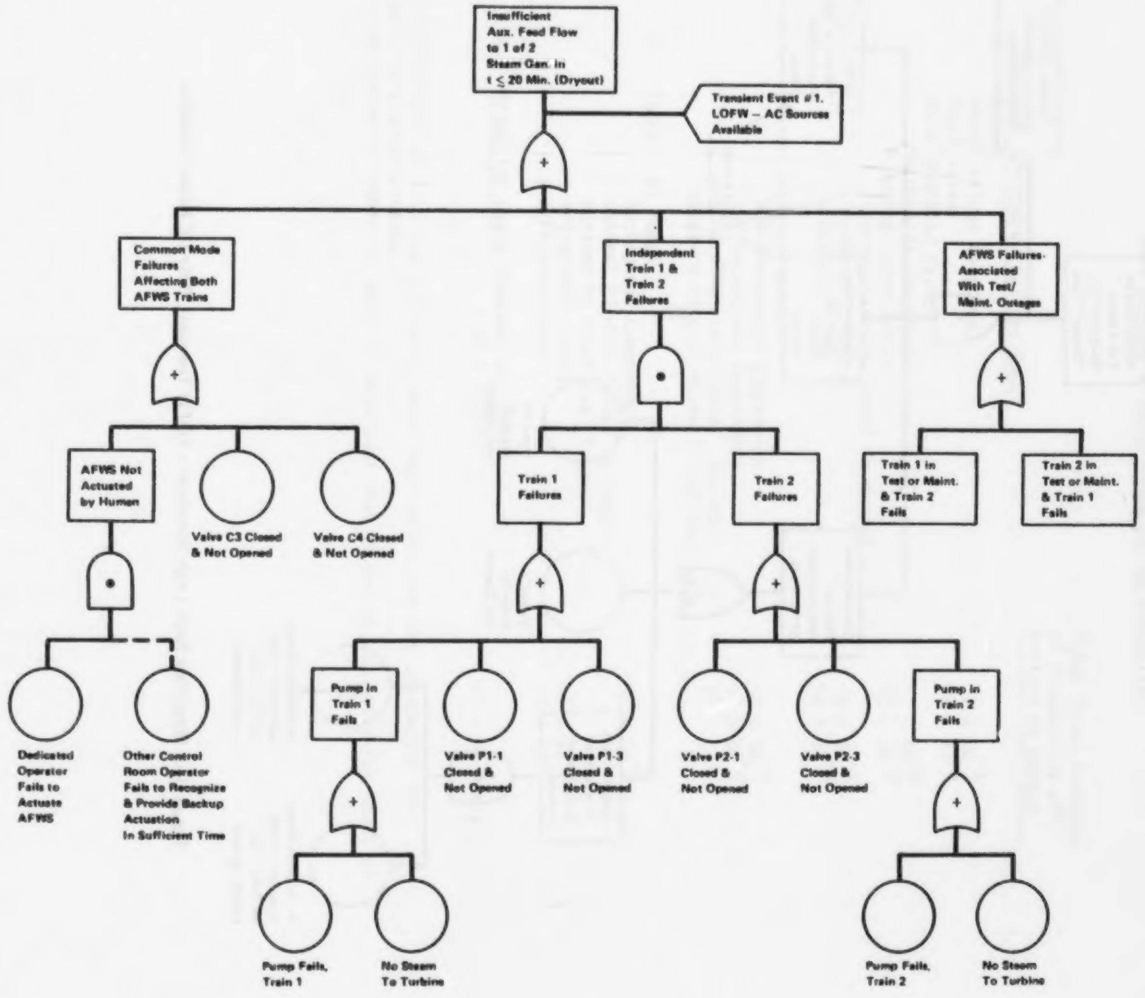
*Evaluating the variability in AFWS designs was the principal aim in this assessment rather than evaluating variability in data to be applied to a specific design.

Figure III-2 Simplified Fault Tree Logic Structure – LOFW Transient.


Insufficient Aux. Feed Flow to 1 of 2 Steam Gen. in t ≤ 20 Min. (Dryout)
Transient Event # 1. LOFW – AC Sources Available
+
Common Mode Failures Affecting Both AFWS Trains
Independent Train 1 & Train 2 Failures
AFWS Failures Associated With Test/ Maint. Outages
+
AFWS Not Actuated by Human
Valve C3 Closed & Not Opened
Valve C4 Closed & Not Opened
•
Dedicated Operator Fails to Actuate AFWS
Other Control Room Operator Fails to Recognize & Provide Backup Actuation In Sufficient Time
•
Train 1 Failures
Train 2 Failures
+
Pump in Train 1 Fails
Valve P1-1 Closed & Not Opened
Valve P1-3 Closed & Not Opened
+
Pump Fails, Train 1
No Steam To Turbine
+
Valve P2-1 Closed & Not Opened
Valve P2-3 Closed & Not Opened
Pump in Train 2 Fails
+
Pump Fails, Train 2
No Steam To Turbine
+
Train 1 in Test or Maint. & Train 2 Fails
Train 2 in Test or Maint. & Train 1 Fails

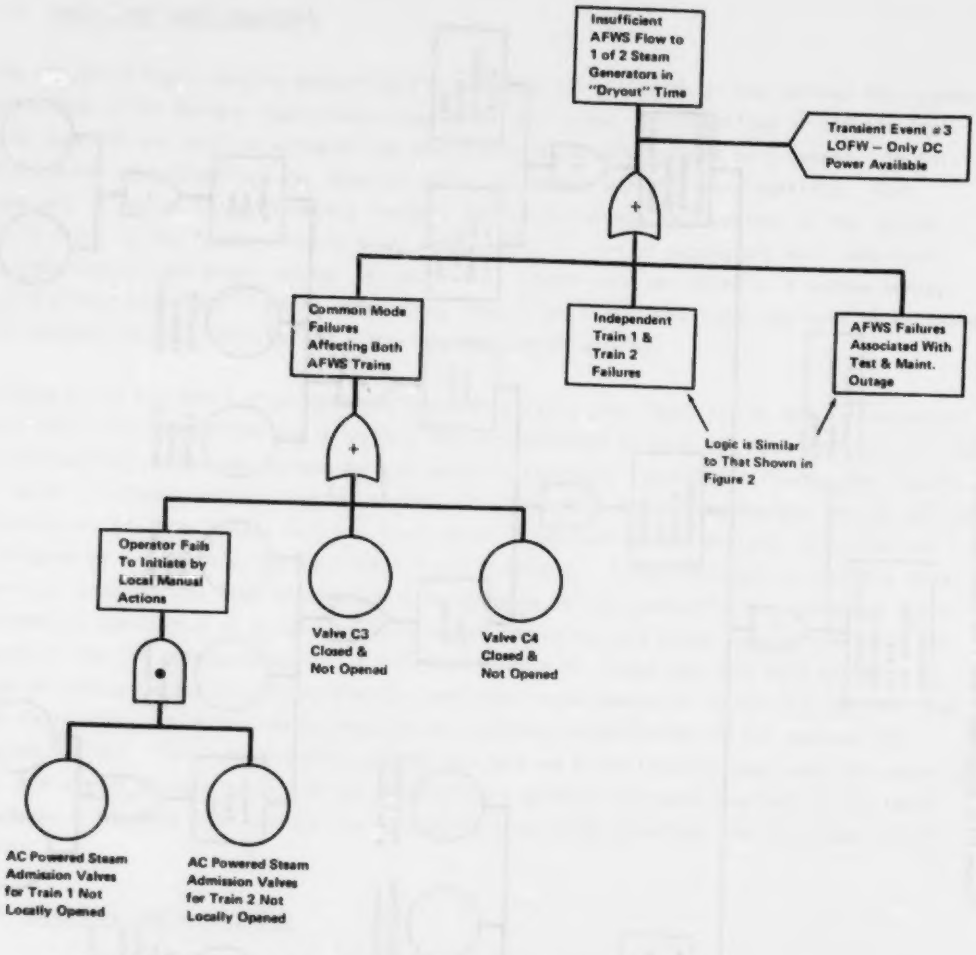

Insufficient AFWS Flow to 1 of 2 Steam Generators in "Dryout" Time
Transient Event #3 LOFW – Only DC Power Available
+
Common Mode Failures Affecting Both AFWS Trains
Independent Train 1 & Train 2 Failures
AFWS Failures Associated With Test & Maint. Outage
Logic is Similar to That Shown in Figure 2
+
Operator Fails To Initiate by Local Manual Actions
Valve C3 Closed & Not Opened
Valve C4 Closed & Not Opened
•
AC Powered Steam Admission Valves for Train 1 Not Locally Opened
AC Powered Steam Admission Valves for Train 2 Not Locally Opened

Figure III-3 Simplified Fault Logic Structure – LOFW Transient, Only DC Power Available.

TABLE III-2

BASIC DATA USED FOR PURPOSES OF CONDUCTING A COMPARATIVE ASSESSMENT OF EXISTING AFWS DESIGNS & THEIR POTENTIAL RELIABILITIES

	Point Value Estimate of Probability of* Failure on Demand
I. Component (Hardware) Failure Data	
a. Valves:	
Manual Valves (plugged)	$\sim 1 \times 10^{-4}$
Check Valves	$\sim 1 \times 10^{-4}$
Motor Operated Valves	
· Mechanical Components	$\sim 1 \times 10^{-3}$
· Plugging Contribution	$\sim 1 \times 10^{-4}$
· Control Circuit (local to Valve)	
w/quarterly tests	$\sim 6 \times 10^{-3}$
w/monthly tests	$\sim 2 \times 10^{-3}$
Piston Actuated Valves	
· MOV-Mechanical Components	$\sim 3 \times 10^{-4}$
· SOV-Mechanical Components	$\sim 1 \times 10^{-3}$
· Control Circuit (Note: Use MOV Failure Rate if Valve is not Fail Safe)	ε**
b. Pumps: (1 Pump)	
· Mechanical Components	$\sim 1 \times 10^{-3}$
· Control Circuit (Local to Pump - applies to Electrical Pumps)	
w/Quarterly tests	$\sim 7 \times 10^{-3}$
w/Monthly tests	$\sim 4 \times 10^{-3}$
c. Actuation Logic (Assumes at least 1 of 2 logic)	$\sim 7 \times 10^{-3}$/train

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*Error factors of 3-10 (up and down) about such valves are not unexpected for basic data uncertainties.

**ε represents a number so small in magnitude that it may be neglected for basis of this study.

TABLE III-2 (Continued)

II. TEST & MAINTENANCE OUTAGE CONTRIBUTIONS:

a. Calculational Approach

1. Test Outage

$$Q_{TEST} \approx \frac{(\#hrs/test)\ (\#tests/year)}{\#hrs/year}$$

2. Maintenance Outage

$$Q_{MAINT.} \approx \frac{0.22\ (\#hrs/maint.\ act)}{720}$$

b. Data Tables for Test & Maint. Outages*

SUMMARY OF TEST ACT DURATION

Component	Range on Test Act Duration Time, hr	Calculated Mean Test Act Duration Time, t_D, hr
Pumps	0.25 - 4	1.4
Valves	0.25 - 2	0.86
Diesels	0.25 - 4	1.4
Instrumentation	0.25 - 4	1.4

LOG-NORMAL MODELED MAINTENANCE ACT DURATION

Component	Range On Duration Time, hr	Mean Act Duration Time, hr
Pumps	1/2 - 24	7
	1/2 - 72	19
Valves	1/2 - 24	7
Diesels	2 - 72	21
Instrumentation	1/4 - 24	6

---

* **Note**: These data tables were taken from the Reactor Safety Study (WASH-1400)(2) for purposes of this AFW system assessment. Where the plant technical specifications placed limits on the outage duration(s) allowed for AFW system trains, this tech spec limit was used to estimate the mean duration times for maintenance ACPS. In general, it was found that the outages allowed for maintenance dominated those contributions to AFW system unavailability from outages due to testing.

TABLE III-2 (Continued)

III. Human Acts & Errors - Failure Data:

	Estimated Human Error/Failure Probabilities Modifying Factors & Situations					
	With Valve Position Indication in Control Room		With Local Walk-Around & Double Check Procedures		W/O Either	
	Point Value Est	Est. on Error Factor	Point Value Est	Est. on Error Factor	Point Value Estimate	Est On Error Factor
a. Acts & Errors of A Pre-Accident Nature						
1. Valves Mispositioned During Test/Maint						
(a) Specific Single Valve Wrongly Selected out of A Population of Valves During Conduct of a Test or Maintenance Act (X No. of Valves in Population at Choice)	$\frac{1}{20}$ " 10^{-2} " $\frac{1}{X}$	20	$\frac{1}{2}$ " 10^{-2} " $\frac{1}{X}$	10	10^{-2} " $\frac{1}{X}$	10
(b) Inadvertently Leaves Correct Valve in Wrong Position	N5 x 10^{-4}	20	N5 x 10^{-3}	10	N10^{-2}	10
2. More than one valve is affected (coupled errors)	N1 x 10^{-4}	20	N1 x 10^{-3}	10	N3 x 10^{-3}	10
3. Miscalibration of Sensors/Electrical Relays						
(a) One Sensor/Relay Affected	-	-	N5 x 10^{-3}	10	N10^{-2}	10
(b) More than one Sensor/Relay Affected	-	-	N1 x 10^{-3}	10	N3 x 20^{-3}	10

TABLE III-2 (Continued)

	Time Actuation Needed	Estimated Failure Prob. for Primary Operator to Actuate AFWS	Estimated Failure Prob. of other (Backup) Control Rm. Operator to Actuate AFWS	Overall Estimate of Failure Probability	Estimated Error Factor on Overall Probability
b. Acts & Errors of a Post-Accident Nature					
1. Manual Actuation of AFW system from Control Room					
(a) Considering "Dedicated" Operator to Actuate AFW system and Possible Backup Actuation of AFWS	N5 min.	N2 x 10^{-3}	-	N2 x 10^{-3}	10
	N15 min.	N1 x 10^{-3}	N0.5 (mod. dep.)	N5 x 10^{-4}	10
	N30 min.	N5 x 10^{-4}	N.25 (low dep.)	N10^{-4}	10
(a) Considering "Non-Dedicated" Operator to Actuate AFW system and Possible Backup Acutation of AFW system	N5 min.	N5 x 10^{-2}	-	N5 x 10^{-2}	10
	N15 min.	N1 x 10^{-2}	N0.5 (mod. dep.)	N5 x 10^{-3}	10
	N30 min.	N5 x 10^{-3}	N.25 (low dep.)	N10^{-3}	10

Study and from discussions with recognized experts in the field of human behavior and reliability at Sandia Laboratories. The Sandia experts are presently working with NRC's Office of Nuclear Regulatory Research on human factors-related programs.

The best estimate data in Table III-2 are subject to considerable uncertainty, and may have error spreads of an order of magnitude on either side of the data. Hence, although the data base may be used to obtain relative reliabilities, care must be taken in ascribing a high degree of numerical precision to these values (5) or to results derived from their use. Because of this, any relative values for AFW system reliability presented herein should not be interpreted to have a high degree of precision. The data from Table III-2 were applied* to the fault logic structure in order to obtain relative comparisons of reliability of the various AFW system designs. It was found that the various AFW system designs did exhibit considerable variability with regard to design approach and in their human influences. For example, some AFW systems include three feedwater pumps (two electric motor-driven and one steam turbine-driven), are automated, and no single point vulnerability was identified in our review. In contrast, some AFW systems have two pumps and are not automated, thereby having a strong dependence on human influences for their performance. In addition, some of these designs also have single point vulnerabilities that could potentially negate the two train AFW system redundancy (e.g., a single manual valve). Clearly, one might reasonably expect to find a significant variance in reliability between such designs without having available an abundance of data of great precision.

4.6 Summary of Reliability-Based Results (Generic)

Figure III-4 illustrates the results of the generic AFW system reliability assessment. As can be seen from Figure III-4, preliminary assessments of the reliabilities of the AFW system designs range from high to low. On a more quantitative basis, this range depicts differences in reliability of the existing AFW systems of more than an order of magnitude for each of the three transients considered in this assessment. Each column in Figure III-4 depicts the relative reliability of the various system designs for a particular transient.

Plant-specific and generic recommendations to improve on and strengthen AFW system reliability were developed as part of this overall study and are presented in Section 5 of this appendix. These recommendations reflect the engineering insights derived from this reliability evaluation as well as those derived from the deterministic evaluation. The

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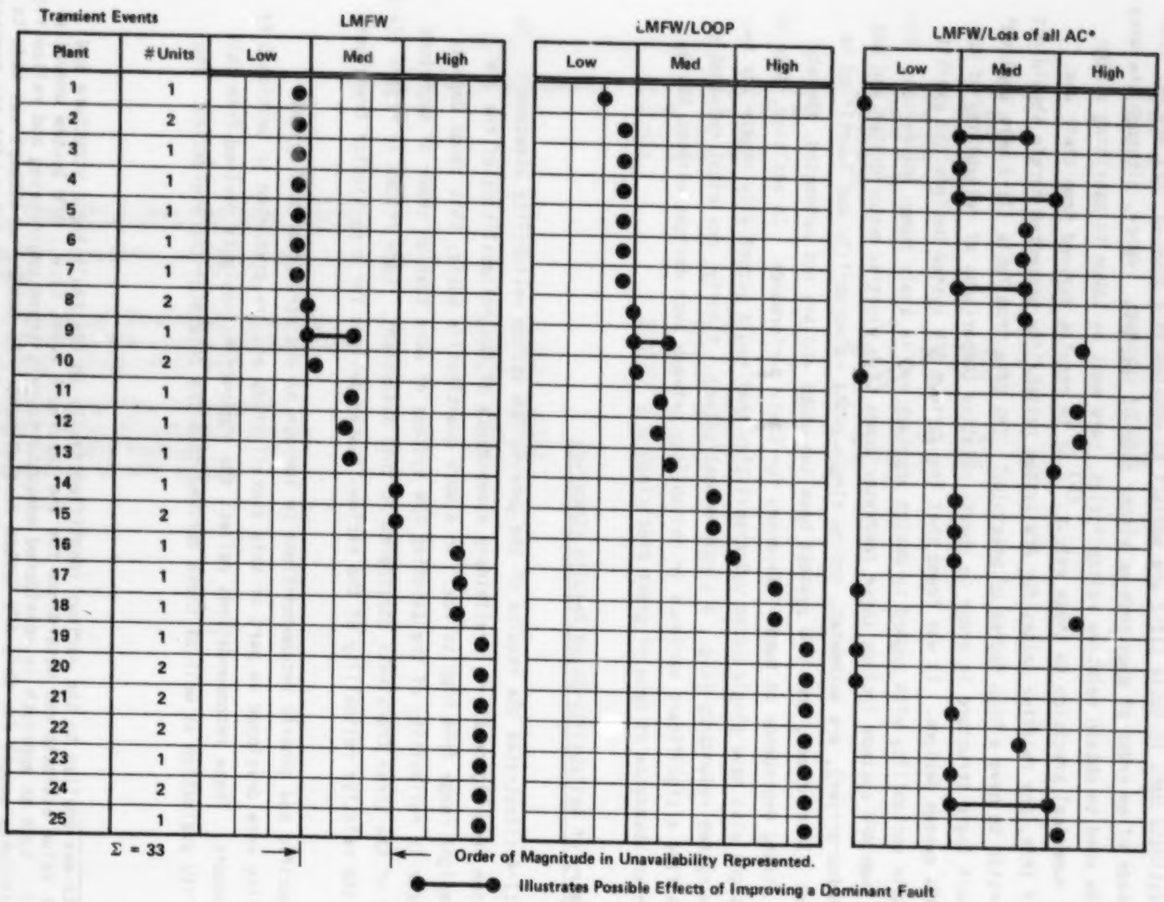
*The data was applied to the various identified faults in the fault logic structure and a point value estimate was determined for the top fault event (i.e., AFW System unavailability). Such an approach is considered adequate to gain those engineering and reliability based insights sought for this AFW System reassessment. As noted, no attempt was made to introduce the somewhat time consuming, calculational elegance, associated with the process of error propagation into this assessment (e.g., Monte Carlo). Prior experience with such a calculational process has revealed a somewhat predictable outcome that, even with the very redundant systems, could be slightly higher than the point value solution (e.g., factor of approximately three times higher than the point value and usually less). Should there exist a clearly overwhelming fault in a systems design, then the process of error propagation would be expected to be merely one of higher elegance and it would yield no important change to the quantitative solution.



Plant	# Units
1	1
2	2
3	1
4	1
5	1
6	1
7	1
8	2
9	1
10	2
11	1
12	1
13	1
14	1
15	2
16	1
17	1
18	1
19	1
20	2
21	2
22	2
23	1
24	2
25	1
	$\Sigma = 33$

*Note. The scale for this event is not the same as that for the LMFW and LMFW/LOOP.

Figure III-4 A Generic Perspective: Comparisons of AFWS Reliability.

recommendations derived from the reliability evaluation generally tend to reduce human error potentials and other dominant failures, and are proposed for all AFW system designs as applicable for all of the AFW system designs regardless of whether these designs are characterized as having a relatively high or low reliability. The three transients used in the assessment are described in detail below.

4.6.1 Loss of Main Feedwater (First Column, Figure III-4)

Approximately eight units were identified as having AFW systems with relatively low reliability for this transient. These AFW system designs generally require manual actuation and include two pumps in their design. Some were found to have single point vulnerabilities such as a single manual valve through which all AFW flow passes (typically a maintenance valve), where human error possibility was generally found to be the dominant common mode failure contributor. In some cases, deficiencies in Technical Specifications are the principal contributor to unavailability; e.g., limits were not imposed on the allowed outage interval for an AFW system train. Where such a deficiency was identified, the reliability of the AFW system could be adversely affected if one of the trains was to be allowed to be inoperable for an extended period of time. In general, for the eight units characterized as having a relatively low AFW system reliability for this transient, the dominant failure is the failure to manually initiate the AFW system. Plants requiring manual AFW system initiation are presently required, by recent IE Bulletins(9,10) to provide a dedicated individual to manually actuate the AFW system upon loss of main feedwater. The results presented in Figure III-4 consider only the reliability of this dedicated individual to actuate the AFW system. It is likely that, in the event that this dedicated individual fails to perform the AFW system actuation, backup would be provided by licensed reactor operators in the control room. Discussions with experts on human reliability indicate that the chance of failing to actuate the AFW system from the control room might be reduced by a factor of two to four by the backup operator, depending on the time window available (see data tables). If this potential for improved human reliability were to be factored into the Figure III-4 results, then other potential failures, such as the single valve vulnerabilities, could become the dominant contributors to the unavailability of AFW systems. Therefore, the net benefit in AFW system unavailability might be limited to the aforementioned factor of two, unless the next level of dominant failure modes were to be improved upon. The degree to which such successive improvements might further improve AFW system reliability was beyond the present work scope. However, recommendations are made in Section 5 of this appendix that should improve these next levels of dominating faults, such as the single manual valve.

Those AFW system designs that could be characterized as being of medium reliability generally were automatically actuated with manual backup. However, single point vulnerabilities were identified which would limit the reliability. Other factors, such as the lack of specific limitations on allowed AFW train outage time and limitations on AFW flow rate to the steam generators because of water-hammer concerns, could have an adverse effect on the AFW system redundancy and thus limit the achievable reliability. Improvements in these areas would serve to further improve reliability of these AFW system designs.

Approximately 16 units were characterized as having high AFW system reliability. These AFW system designs were generally of high redundancy and had no observed single point vulnerabilities. Consequently, the reliability of these designs would be expected to be limited by human interactions that could adversely affect the installed hardware redundancy. For example, some periodic tests of AFW systems are conducted in ways that could invalidate AFW system redundancy. Usually, such tests are not staggered (i.e., each redundant component of the AFW system is tested by the same personnel and on the same shift) such that if identical human errors were to be made on redundant components the entire AFW system could be made ineffective. The net result is that the effect of these human errors could persist until the next test interval, at which time the errors should be detected. To reduce such potential vulnerabilities, recommendations were made for strengthened administrative controls (e.g., improved valve locking procedures) and considerations are being given for staggering tests of the individual AFW system trains, such that only one train would be tested on any given shift. Additional insights derived from this evaluation suggest that the quality of periodic testing, as well as of the AFW system design, should be improved. For example, testing programs that incapacitate more than one train of the AFW system should be revised so that the periodic tests demonstrate availability of flow path to the steam generators rather than negate the flow path.

4.6.2 Loss of Main Feedwater Due to Loss of Offsite Power (Second Column, Figure III-4)

The reliabilities of the various AFW system designs for this transient were generally found to be quite similar to those for the previous transient, i.e., loss of main feedwater. Onsite ac power sources were considered and the potential impact of degrading these power sources (e.g., the loss of one of the two emergency diesel-generators) on the AFW system reliability was estimated. Depending on the AFW system design and on the ac power dependencies identified, variable impacts were estimated. However, these variations generally were not dominant failure modes, and were similar to those previously described for the loss of main feedwater transient.

4.6.3 Loss of Main Feedwater and Loss of All Alternating Current Power (Station ac Blackout), (Third Column, Figure 4)

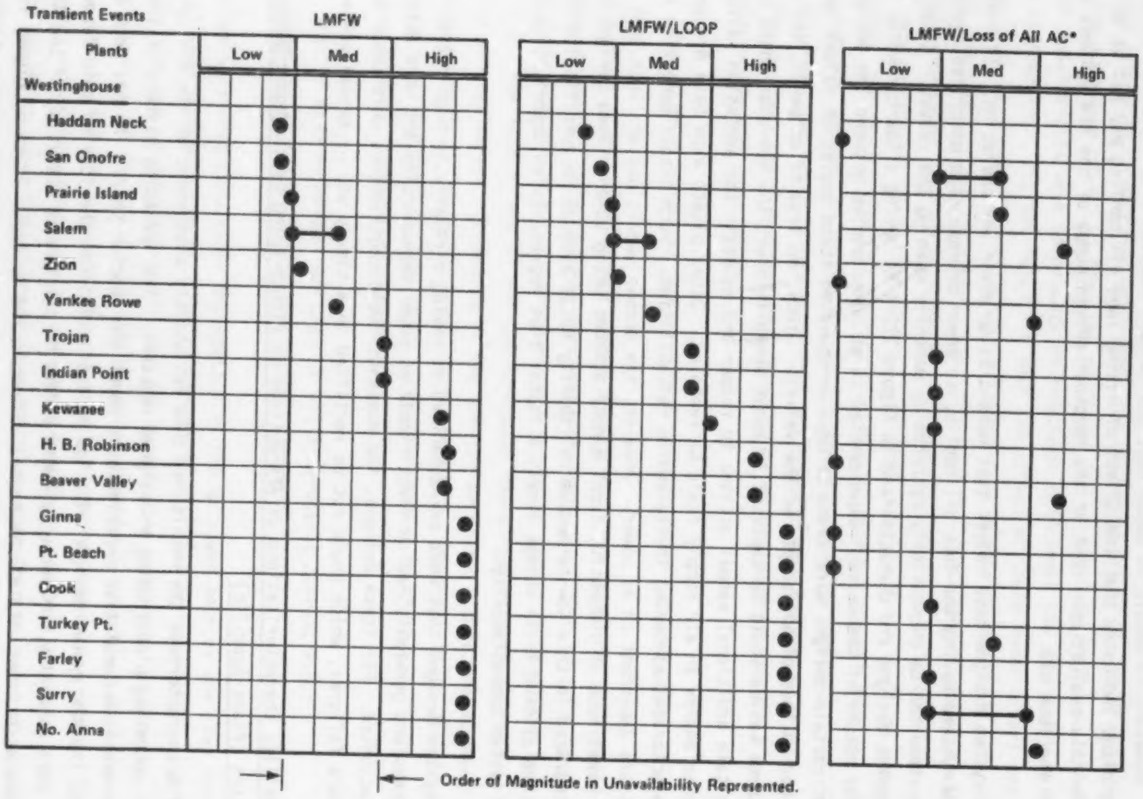
This assessment carried postulated degradation of the ac power sources one step further than the loss of main feedwater and the loss of offsite power. All ac power sources were assumed unavailable, and the ac dependencies of the AFW system were explored. In general, the steam turbine-driven pump of the AFW system was the only potentially operable system for this scenario. Some of the AFW system designs have only steam turbine-driven pumps. Therefore, these designs potentially have greater available redundancy for this scenario. The relative reliability of the various AFW system designs varied by more than an order of magnitude for this transient. Seven reactor units were characterized as having relatively low reliabilities for this transient. These particular AFW system designs did not necessarily follow their prior characterizations in Columns 1 and 2 of Figure III-4. This difference is due to the strong ac dependencies which exist in the steam turbine-driven train of their AFW system. All seven units depend on ac power to provide lube oil cooling for the steam turbine-driven pump. Without this lube oil cooling, it was assumed that the pump would overheat and eventually fail due to shaft/seal/bearing failures. Estimates on when pump failure might occur vary, but it was assumed to occur in a relatively short time interval. As a consequence, those AFW systems having this ac power dependency were judged to have a low reliability for this event. However, it should be noted that preliminary results from a subsequent test at an operating plant indicated that the effect of losing lube oil cooling may not be as rapid as assumed in this evaluation. Most of the turbine- driven pumps of the AFW systems use AFW flow to cool the lube oil.

Several AFW system designs have valves that depend on ac power for operation. In contrast to those AFW systems having the lube-oil cooling ac power dependencies described above, these AFW systems can be successfully operated by manually opening the valves. Generally, these AFW system designs are characterized in Figure III-4 as having a low-to-medium reliability. The nature of the valves' dependencies on ac power varied between the designs. For example, certain designs were found to have ac-operated steam admission valves designed to fail closed on loss of air supply to the valves. Since, on loss of ac power, the air supply to these valves could be depleted in about one-half hour, the operator would be required to take additional manual actions to reopen and maintain the admission valves open, until ac power and/or an air supply could be restored. Other plants also have AFW system designs characterized as having low-to-medium reliabilities. Such plants generally include valves that are dependent on ac power. However, the access conditions are such that they reduce the likelihood of successful local manual actions being taken. Some designs were also characterized in this low-to-medium reliability range because no specific limitations existed on the allowed train outage times, a factor that represents an important contributor to the AFW system unavailability.

Those AFW system designs that were characterized as having a relatively high reliability for this transient generally had no identifiable ac power dependencies and were automatically actuated. For these designs, the dominant fault contributors were those associated with hardware failures, which could not be rectified in a timely way by manual actions.

4.7 Reliability Characterizations of AFW Systems in Plants Using Westinghouse-Designed Reactors (Plant-Specific)

Figure III-5 characterizes the results of this reliability assessment of the AFW system designs in operating plants using W-designed reactors. The operating history of these plants represents a cumulative experience of about 150 reactor years. As has been discussed previously, the AFW system designs with low reliabilities for the loss of main feedwater transient were generally dominated by human errors in the manual actuation of the AFW system, or by errors associated with single manual valves in the system. Those of higher reliability were also generally dominated by human influences, which could affect the redundant aspect of the AFW system design. In general, the majority of these AFW system designs are of a configuration that includes three AFW pumping systems which are not vulnerable to single point failures. Therefore, they were assessed to be of relatively high reliability.



*Note: The scale for this event is not the same as that for the LMFW and LMFW/LOOP.

Figure III-5 Reliability Characterizations for AFWS Designs in Plants Using the Westinghouse NSSS.

In connection with the loss of main feedwater transient and the loss of all ac power, there is a significant variation in the reliability of the various AFW system designs. This variation is attributed to differences in the design of the AFW systems previously discussed. Those plants having an ac power dependency (e.g., lube oil cooling to the steam turbine-driven AFW pump) have the lowest AFW system reliability for this transient. Accordingly, recommendations are made to eliminate power dependencies which could result in pump failure within a short time interval.

Our assessment of each of the plants listed in Figure III-5 is described in Appendix X.

The reliability assessment approach used and the principal insights and results are summarized below. The results shown in Figures III-4 and III-5 indicate that the reliabilities of the existing AFW system designs vary by at least an order of magnitude. The dominant contributors to this variability in reliability were, in general, human errors and single point vulnerabilities, as described later in this appendix.

5. SUMMARY OF RESULTS AND RECOMMENDATIONS

This section presents in summary form the results of the AFW system review and recommendations that should be implemented to improve the performance and reliability of the AFW systems of the various W-designed operating plants. Appendix X provides a separate AFW system description, evaluation, and recommendations for each individual plant.

5.1 Recommendation Categories

The recommendations are categorized as generic and plant-specific, as well as short-term and long-term. The generic recommendations (designations GS and GL refer to generic short- and long-term, respectively) are a result of similarities in AFW system potential problems between plants and are applicable to more than one plant. The generic recommendations and the concerns which led to these recommendations are described in this section. There are also plant-specific recommendations that are unique to a given plant's AFW system. The plant-specific recommendations are addressed more fully in the individual plant evaluations in Appendix X.

The short-term recommendations represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable. This implementation schedule is intended to be consistent with the schedule for implementation of the requirements specified in NUREG-0578 (13). If conflicts should arise; the schedule specified in NUREG-0578 takes precedence.

There are two significant limitations of the AFW system review and evaluation which should be noted, as well as their effect on the recommendations.

(1) While our review covered the classification and divisional redundancy of power sources for AFW system equipment and instrumentation and controls and the type of instrumentation and controls provided for the overall AFW system, we did not attempt to review detailed logic and control diagrams. This explains in part the conservative approach we used in applying to all plants the short and long-term generic recommendations GS-7 and GL-5, which deal with non-redundant and non-Class 1E circuitry for AFW system automatic initiation systems.

(2) The review is not considered to be a complete evaluation of postulated high energy pipe breaks that could affect the AFW system, since piping isometric and plant arrangement drawings were not reviewed. However, where system flow sheets revealed potential pipe breaks that could cause total loss of AFW system capability, these problem areas have been identified and included in the long-term recommendations for further evaluation.

5.2 Short-Term Generic Recommendations

5.2.1 Technical Specification Time Limit on AFW System Train Outage

Concern

Several of the plants reviewed have Technical Specifications that permit one of the AFW system trains to be out of service for an indefinite time period. Indefinite outage of one train reduces the defense-in-depth provided by multiple AFW system trains.

Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

5.2.2 Technical Specification Administrative Controls on Manual Valves - Lock and Verify Position

Concern - Several of the plants reviewed use a single manual valve or multiple valves in series in the common suction piping between the primary water source and the AFW system pump suction. At some plants the valves are locked open, while at others, they are not locked in position. If the valves are inadvertently left closed, the AFW system would be inoperable, because the water supply to the pumps would be isolated. Since there is no remote valve position indication for these valves, the operator has no immediate means of determining valve position.

Further, the Technical Specifications for plants with locked-open manual valves do not require periodic inspection to verify that the valves are locked and in the correct position. For most plants where the valves are not locked open, valve position is verified on some periodic basis.

Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

5.2.3 AFW System Flow Throttling-Water Hammer

Concern - Several of the plants reviewed apparently throttle down the AFW system initial flow to eliminate or reduce the potential for water hammer. In such cases, the overall reliability of the AFW system can be adversely affected.

Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

5.2.4 Emergency Procedures for Initiating Backup Water Supplies

Concern - Most of the plants do not have written procedures for transferring to alternate sources of AFW supply if the primary supply is unavailable or exhausted. Without specific criteria and procedures for an operator to follow to transfer to alternate water sources, the primary supply could be exhausted and result in pump damage or a long interruption of AFW flow.

Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

(1) The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated.

(2) The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

5.2.5 Emergency Procedures for Initiating AFW Flow Following a Complete Loss of Alternating Current Power

Concern - Some operating plants depend on ac power for all sources of AFW system supply, including the turbine-driven pump train. In the event of loss of offsite and onsite ac power, ac-dependent lube oil supply or lube oil cooling for the pump will stop, and/or manual actions are required to initiate AFW flow from the turbine-driven pump by manually opening the turbine steam admission valve and/or AFW system flow control valves. There are no procedures available to the plant operators for AFW system initiation and control under these conditions. This could result in a considerable time delay for AFW system initiation, since the operators would not be guided by procedures dealing with this event.

Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train, independent of any ac power source. If manual AFW system initiation or flow control is required following a complete loss of ac power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on ac power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all ac power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until ac power is restored. Adequate lighting powered by direct current (dc) power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer term resolution of this concern.)

5.2.6 AFW System Flow Path Verification

Concern - Periodic testing of the AFW system is accomplished by testing of individual components of one flow train (periodic pump recirculation flow test or automatic valve actuation), thus altering the normal AFW system flow path(s). The flow capability of the entire AFW system, or at least one integral AFW system train, is only demonstrated on system demand following a transient, or if the AFW system is used for normal plant startup or shutdown.

Recent Licensee Event Reports indicate a need to improve the quality of system testing and maintenance. Specifically, periodic testing and maintenance procedures inadvertently result in (1) more than one AFW system flow train being unavailable during the test, or (2) the AFW system flow train under test not being properly restored to its operable condition following the test or maintenance work. The Office of Inspection and Enforcement has taken action to correct Item (1); the recommendation below is made to correct Item (2).

Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

(1) Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

(2) The licensee should propose Technical Specifications to assure that, prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5.2.7 Non-Safety Grade, Non-Redundant AFW System Automatic Initiation Signals

Concern - Some plants with an automatically initiated AFW system utilize some initiation signals that are not safety-grade, do not meet the single failure criterion, and are not required by the Technical Specifications to be tested periodically. This can result in reduced reliability of the AFW system.

Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer-term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements, as indicated in Recommendation GL-5.

(1) The design should provide for the automatic initiation of the AFW system flow.

(2) The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of AFW system function.

(3) Testability of the initiation signals and circuits shall be a feature of the design.

(4) The initiation signals and circuits should be powered from the emergency buses.

(5) Manual capability to initiate the AFW system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

(6) The ac motor-driven pumps and valves in the AFW system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

(7) The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

5.2.8 Automatic Initiation of AFW Systems

Concern - For plants with a manually initiated AFW system, there is the potential for failure of the operator to manually actuate the system following a transient in time to maintain the steam generator water level high enough to assure reactor decay heat removal via the steam generator(s). While IE Bulletin 79-06A requires a dedicated individual for W-designed operating plants with a manually initiated AFW system, further action should be taken in the short-term. This concern is identical to Item 2.1.7a of NUREG-0578.(13)

Recommendation GS-8 - The licensee should install a system to automatically initiate AFW system flow. This system need not be safety-grade; however, in the short-term, it should meet the criteria listed below, which are similar to Item 2.1.7.a of NUREG-0578.(13) For the longer-term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements, as indicated in Recommendation GL-2.

(1) The design should provide for the automatic initiation of the AFW system flow.

(2) The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of AFW system function.

(3) Testability of the initiating signals and circuits should be a feature of the design.

(4) The initiating signals and circuits should be powered from the emergency buses.

(5) Manual capability to initiate the AFW system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

(6) The ac motor-driven pumps and valves in the AFW system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

(7) The automatic initiation signals and circuits should be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

5.3 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins & Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants, subsequent to our review of the AFW system designs at W- and C-E- designed operating plants. They have not been examined for specific applicability to individual W-and CE-designed operating plants.

5.3.1 Primary AFW Water Source Low Level Alarm

Concern - Plants which do not have level indication and alarm for the primary water source may not provide the operator with sufficient information to properly operate the AFW system.

Recommendation - The licensee should provide redundant level indication and low level alarms in the control room for the AFW system primary water supply, to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

5.3.2 AFW Pump Endurance Test

Concern - Since it may be necessary to rely on the AFW system to remove decay heat for extended periods of time, it should be demonstrated that the AFW pumps have the capability for continuous operation over an extended time period without failure.

Recommendation - The licensee should perform a 72 hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72 hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

5.3.3 Indication of AFW Flow to the Steam Generators

Concern - Indication of AFW flow to the steam generators is considered important to the manual regulation of AFW flow to maintain the required steam generator water level. This concern is identical to Item 2.1.7.b of NUREG-0578.(13)

Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:(13)

(1) Safety-grade indication of AFW flow to each steam generator should be provided in the control room.

(2) The AFW flow instrument channels should be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the AFW system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

5.3.4 AFW System Availability During Periodic Surveillance Testing

Concern - Some plants require local manual realignment of valves to conduct periodic pump surveillance tests on one AFW system train. When such plants are in this test mode and there is only one remaining AFW system train available to respond to a demand for initiation of AFW system operaiton, the AFW system redundancy and ability to withstand a single failure are lost.

Recommmendation - Licensees witn plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system from the test mode to its operational alignment.

5.4 Long-Term Generic Recommendations

5.4.1 Automatic Initiation of AFW Systems

Concen - This concern is the same as short-term generic recommendation GS-8; namely, failure of an operator to actuate a manual start AFW system in time to maintain steam generator water level high enough to assure reactor decay heat removal via the steam generator(s).

Recommendation GL-1 - For plants with a manual starting AFW system, the licensee should install a system to automatically initiate the AFW system flow. This system and associated automatic initiation signals should be designed and installed to meet safety-grade requirements. Manual AFW system start and control capability should be retained with manual start serving as backup to automatic AFW system initiation.

5.4.2 Single Valves in the AFW System Flow Path

Concern - This concern is the same as short-term generic recommendation GS-2 -- namely, AFW system inoperability due to an inadvertently closed manual valve that could interrupt all AFW system flow.

Recommendation GL-2 - Licensees with plant designs in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

Licensees with plant designs in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions into the surveillance requirements.

5.4.3 Elimination of AFW System Dependency on Alternating Current Power Following A Complete Loss of Alternating Current Power.

Concern - This concern is the same as short-term generic recommendation GS-5 - namely, delay in initiation of AFW system operation or maintaining AFW system operation following a postulated loss of onsite and offsite ac power; i.e., ac power blackout.

Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any ac power source for at least two hours. Conversion of dc power to ac power is acceptable.

5.4.4 Prevention of Multiple Pump Damage Due to Loss of Suction Resulting From Natural Phenomena

Concern - In many of the operating plants, the normal water supply to the AFW system pumps (including the interconnected piping) is not protected from earthquakes or tornadoes. Any natural phenomenon severe enough to result in a loss of the water supply could also be severe enough to cause a loss of offsite power with loss of main feedwater, resulting in an automatic initiation signal to start the AFW system pumps. The pumps would start without any suction head, leading to cavitation and multiple pump damage in a short period of time, possibly too short for the operators to take action that would protect the pumps. This may lead to unacceptable consequences for some plants, due to a complete loss of feedwater (main and auxiliary).

Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure, or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

5.5.5 Non-Safety Grade, Non-Redundant AFW System Automatic Initiation Signals

Concern - This concern is the same as short-term generic recommendation GS-7 - namely, reduced AFW system reliability as a result of use of non-safety-grade, non-redundant signals, which are not periodically tested, to automatically initiate the AFW system.

Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

5.5 Plant Specific AFW System Recommendations

The short-term and long-term plant specific recommendations applicable to the AFW systems for each plant are identified and discussed in Appendix X.

5.6 Summary of AFW System Recommendations for Westinghouse-Designed Operating Plants

Table III-3 below summarizes the short-term and long-term generic and plant-specific recommendations for the AFW system at each W-designed operating reactor. The additional generic short-term recommendations discussed in 5.3 of this appendix are not included in Table III-3. However, these recommendations are included in the individual plant AFW system evaluations contained in Appendix X.

Table III-3 (W)

	Generic Recommendations													Plant Specific Recommendations	
	GS-1:Tech Spec LCO Train Outage Time Limit	GS-2:Tech Spec-Single Flow Path Manual Valves	GS-3:AFWS Flow Throttling-Water Hammer	GS-4:Emergency Procedure-Backup Water Supplies	GS-5:Emergency Procedure-AC Power Blackout	GS-6:Flow Path Verification	GS-7:Non-Safety Grade AFWS Automatic Start Signals	GS-8:Automatic Actuation of AFWS	GL-1:Automatic Actuation of AFWS	GL-2:Single Flow Path-Manual Valves	GL-3:Eliminate AC Power Dependency of One Train	GL-4:Multiple Pump Damage Protection-Natural Phenomena	GL-5:Non-Safety Grade AFWS Automatic Start Signals		
Plant	Short Term								Long Term					Short Term	Long Term
Beaver Valley #1 2-elect pumps 1-turbine pump Automatic Initiation Initiation			X	X		X	X						X	-Modify Tech Specs to require periodically testing and verification of position of manual valves from River Water System to AFWS. -Implement improved procedures for locking manual valves and stagger testing of AFWS pumps. -Re-evaluate alignment of AFW pump discharge block valve to assure required AFW flow for normal, transient, and accident plant conditions.	-Evaluate postulated AFWS discharge pipe break concurrent with single active failure to determine AFW system modifications or procedures necessary or describe how plant can be safely shutdown by use of other available systems.
D.C. Cook 1&2 1 elect pump* 1 turbine pump* *per unit (elect pumps shared) Automatic Initiation		X		X	X	X	X				X	X	X		-Complete licensee proposed modifications to eliminate turbine pump train dependence on AC power.
Farley #1 2 elect pumps 1 turbine pump Automatic Initiation				X	X	X	X				X		X		

Table III-3 (W)

	Generic Recommendations													Plant Specific Recommendations	
	GS-1:Tech Spec LCO Train Outage Time Limit	GS-2:Tech Spec-Single Flow Path Manual Valves	GS-3:AFWS Flow Throttling-Water Hammer	GS-4:Emergency Procedure-Backup Water Supplies	GS-5:Emergency Procedure-AC Power Blackout	GS-6:Flow Path Verification	GS-7:Non-Safety Grade AFWS Automatic Start Signals	GS-8:Automatic Actuation of AFWS	GL-1:Automatic Actuation of AFWS	GL-2:Single Flow Path-Manual Valves	GL-3:Eliminate AC Power Dependency of One Train	GL-4:Multiple Pump Damage Protection-Natural Phenomena	GL-5:Non-Safety Grade AFWS Automatic Start Signals		
Plant	Short Term								Long Term					Short Term	Long Term
Ginna Main AFWS 2 elect pumps 1 turbine pump Automatic Initiation Standby AFWS 2 elect pumps Manual Initiation			X		X	X	X				X		X		-Evaluate capability to provide water supply for 2 hours without dependence on AC power. -Complete SEP evaluation of the need for automatic termination of flow to a depressurized steam generator and automatically providing flow to an intact steam generator and capability of (M) and (SB) AFWS to withstand missiles - internal and tornado, flooding, pipe whip, and seismic events.
Haddam Neck 2-turbine pumps Manual Initiation	X	X		X	X	X		X		X	X		X	-Revise monthly test procedure 5.1-13 to eliminate blocking flow path of both pumps -Monthly test procedure 5.1-14 should not be performed with reactor at power	-Revise design to protect against non-isolable break in common header to SG's. -Review MFWS and AFWS capability of withstand pipe break and single active failure. -Complete licensee proposed modifications to eliminate turbine pump train dependence on AC power for lube oil cooling. -Complete SEP evaluation of (a) missiles, pipe whip, earthquakes, flooding and (b) need for automatic termination of flow of a depressured steam generator and...

Table III-3 (W)

Plant	Generic Recommendations													Plant Specific Recommendations	
	Short Term								Long Term					Short Term	Long Term
	GS-1:Tech Spec LCO Train Outage Time Limit	GS-2:Tech Spec-Single Flow Path Manual Valves	GS-3:AFWS Flow Throttling-Water Hammer	GS-4:Emergency Procedure-Backup Water Supplies	GS-5:Emergency Procedure-AC Power Blackout	GS-6:Flow Path Verification	GS-7:Non-Safety Grade AFWS Automatic Start Signals	GS-8:Automatic Actuation of AFWS	GL-1:Automatic Actuation of AFWS	GL-2:Single Flow Path-Manual Valves	GL-3:Eliminate AC Power Dependency of One Train	GL-4:Multiple Pump Damage Protection-Natural Phenomena	GL-5:Non-Safety Grade AFWS Automatic Start Signals		
H.B. Robinson **2-turbine** pumps **1-elect** pump Automatic Initiation	X	X		X	X	X	X			X	X		X	-Modify Tech Specs to require testing of normally locked closed service water and deep well manual valves. -Modify Tech Specs to require monthly testing of turbine pump steam admission valves.	-Evaluate postulated tornado considering single active failure to determine AFW system modifications of procedures to assure AFW water supply or demonstrate how the plant can be safely shutdown.
Indian Point 2 and 3 2-elect pumps* 1-turbine pump* *per unit Automatic Initiation	X	X	X	X		X	X			X			X	-Review effect of loss of air or power to air operated flow control valves and steam admission valves which are powered by same electrical division for Unit #2. Prepare emergency procedure to maintain AFW capability.	-Evaluate capability to withstand pipe break, missiles since all pumps are in one room. -Modify Unit 2 AFW system design to supply power to the controllers of flow control valves and steam admission valves from separate safety grade buses.
Kewaunee 2-elect pumps 1-turbine pump Automatic Initiation	X	X		X		X	X			X		X	X		

Table III-3 (W)

	Generic Recommendations													Plant Specific Recommendations	
	GS-1:Tech Spec LCO Train Outage Time Limit	GS-2:Tech Spec-Single Flow Path Manual Valves	GS-3:AFWS Flow Throttling-Water Hammer	GS-4:Emergency Procedure-Backup Water Supplies	GS-5:Emergency Procedure-AC Power Blackout	GS-6:Flow Path Verification	GS-7:Non-Safety Grade AFWS Automatic Start Signals	GS-8:Automatic Actuation of AFWS	GL-1:Automatic Actuation of AFWS	GL-2:Single Flow Path-Manual Valves	GL-3:Eliminate AC Power Dependency of One Train	GL-4:Multiple Pump Damage Protection-Natural Phenomena	GL-5:Non-Safety Grade AFWS Automatic Start Signals		
Plant	Short Term								Long Term					Short Term	Long Term
No. Anna 1 2-elect pumps* 1-turbine pump* *per unit Automatic Initiation				X		X	X						X		
Prairie Island 1&2 1 elect pump* 1 turbine pump* *per unit elect pump normally feeds opposite unit SG's Automatic Initiation	X	X		X	X	X	X				X	X	X		

Table III-3 (W)

Plant	Generic Recommendations													Plant Specific Recommendations	
	GS-1:Tech Spec LCO Train Outage Time Limit	GS-2:Tech Spec-Single Flow Path Manual Valves	GS-3:AFWS Flow Throttling-Water Hammer	GS-4:Emergency Procedure-Backup Water Supplies	GS-5:Emergency Procedure-AC Power Blackout	GS-6:Flow Path Verification	GS-7:Non-Safety Grade AFWS Automatic Start Signals	GS-8:Automatic Actuation of AFWS	GL-1:Automatic Actuation of AFWS	GL-2:Single Flow Path-Manual Valves	GL-3:Eliminate AC Power Dependency of One Train	GL-4:Multiple Pump Damage Protection-Natural Phenomena	GL-5:Non-Safety Grade AFWS Automatic Start Signals		
	Short Term								Long Term					Short Term	Long Term
Pt. Beach 1&2 1-elect pump* 1-turbine pump* *per unit (elect pumps shared) Automatic Initiation	X	X		X	X	X	X				X	X	X		
Salem 1 2-elect pumps 1-turbine pump Automatic Initiation		X	X	X		X	X			X			X		

Table III-3 (W)

Plant	Generic Recommendations													Plant Specific Recommendations	
	Short Term								Long Term					Short Term	Long Term
	GS-1:Tech Spec LCO Train Outage Time Limit	GS-2:Tech Spec-Single Flow Path Manual Valves	GS-3:AFWS Flow Throttling-Water Hammer	GS-4:Emergency Procedure-Backup Water Supplies	GS-5:Emergency Procedure-AC Power Blackout	GS-6:Flow Path Verification	GS-7:Non-Safety Grade AFWS Automatic Start Signals	GS-8:Automatic Actuation of AFWS	GL-1:Automatic Actuation of AFWS	GL-2:Single Flow Path-Manual Valves	GL-3:Eliminate AC Power Dependency of One Train	GL-4:Multiple Pump Damage Protection-Natural Phenomena	GL-5:Non-Safety Grade AFWS Automatic Start Signals		
San Onofre #1 1 elect pump 1 turbine pump Manual Initiation		X		X	X	X		X	X	X	X	X		-Periodically test normally closed manual valve -Install operators on normally closed manual discharge valves -Connect fire hose or temporary piping to AFWS suction header	-Evaluate main feed and main steam line breaks in AFW discharge lines concurrent with sincle active failure and determine (1) necessary AFWS design changes including environmental qualification and procedures or (2) demonstrate how plant can be brought to safe shutdown by use of other available systems. -Complete SEP evaluation of (1) capability of the AFW system to withstand tornado and internal missiles, flooding, pipe whip, and seismic events, and (2) the need for automatic termination of flow to a depressurized steam generator and automatically providing flow to an intact steam generator.

Table III-3 (W)

Plant	Generic Recommendations: Short Term: GS-1:Tech Spec LCO Train Outage Time Limit	GS-2:Tech Spec-Single Flow Path Manual Valves	GS-3:AFWS Flow Throttling-Water Hammer	GS-4:Emergency Procedure-Backup Water Supplies	GS-5:Emergency Procedure-AC Power Blackout	GS-6:Flow Path Verification	GS-7:Non-Safety Grade AFWS Automatic Start Signals	GS-8:Automatic Actuation of AFWS	Long Term: GL-1:Automatic Actuation of AFWS	GL-2:Single Flow Path-Manual Valves	GL-3:Eliminate AC Power Dependency of One Train	GL-4:Multiple Pump Damage Protection-Natural Phenomena	GL-5:Non-Safety Grade AFWS Automatic Start Signals	Plant Specific Recommendations: Short Term	Plant Specific Recommendations: Long Term
Surry 1&2 2 elect pumps* 1-turbine pump* *per unit (each AFWS can feed SG's of opposite unit) Automatic Initiation	X			X	X	X	X				X		X	-Lock open manual valves inside containment and periodically verify position. -Stagger periodic testing of pump train tests. -Prepare emergency procedure for operating AFWS of one unit to supply SG's of opposite unit.	-Re-evaluate AFWS design with MOV's inside containment considering environmental induced failure of MOV's or AFWS line break downstream of MOV's.
Trojan 1-diesel pump 1 turbine pump Automatic Initiation		X		X	X	X	X			X	X	X	X		-Evaluate need for planned third pump to be safety grade and installed so that AFW can withstand pipe break plus single active failure.

Table III-3 (W)

	Generic Recommendations													Plant Specific Recommendations	
	GS-1:Tech Spec LCO Train Outage Time Limit	GS-2:Tech Spec-Single Flow Path Manual Valves	GS-3:AFWS Flow Throttling-Water Hammer	GS-4:Emergency Procedure-Backup Water Supplies	GS-5:Emergency Procedure-AC Power Blackout	GS-6:Flow Path Verification	GS-7:Non-Safety Grade AFWS Automatic Start Signals	GS-8:Automatic Actuation of AFWS	GL-1:Automatic Actuation of AFWS	GL-2:Single Flow Path-Manual Valves	GL-3:Eliminate AC Power Dependency of One Train	GL-4:Multiple Pump Damage Protection-Natural Phenomena	GL-5:Non-Safety Grade AFWS Automatic Start Signals		
Plant	Short Term								Long Term					Short Term	Long Term
Turkey Pt. 3&4 3-turbine pumps for both units one pump normally supplies each unit 3rd pump shared backup Automatic Initiation	X	X		X	X	X	X				X		X		-Evaluate postulated pipe break in the single flow path portion of the AFW pump discharge or turbine steam supply line (1) to determine necessary change to AFW design or procedures, or (2) describe how plant can be safely shutdown by use of other available systems -Evaluate shared source of cooling water (city water system) for turbine driven pump lube oil to eliminate commode failures -Provide capability for manual operation of AFWS from the control room -Evaluate break in MFW header upstream of AFW control valve to preclude loss of flow to all steam generators -Reevaluate circuit logic that isolates turbine pump steam supply line on CIS.

Table III-3 (W)

	Generic Recommendations													Plant Specific Recommendations	
	GS-1:Tech Spec LCO Train Outage Time Limit	GS-2:Tech Spec-Single Flow Path Manual Valves	GS-3:AFWS Flow Throttling-Water Hammer	GS-4:Emergency Procedure-Backup Water Supplies	GS-5:Emergency Procedure-AC Power Blackout	GS-6:Flow Path Verification	GS-7:Non-Safety Grade AFWS Automatic Start Signals	GS-8:Automatic Actuation of AFWS	GL-1:Automatic Actuation of AFWS	GL-2:Single Flow Path-Manual Valves	GL-3:Eliminate AC Power Dependency of One Train	GL-4:Multiple Pump Damage Protection-Natural Phenomena	GL-5:Non-Safety Grade AFWS Automatic Start Signals		
Plant	Short Term								Long Term					Short Term	Long Term
Yankee Rowe 1-turbine pump (backed up by charging and safety injection systems) Manual Initiation		X		X	X	X								-Periodically cycle the manual valves that must be operated to connect the charging pumps/safety injection pumps to the AFWS.	-One AFW pump and associated flow path and instrumentation should be capable of initiation and operation from control room for two hours independent of AC power supply. -Evaluate need for charging pumps to be normally supplied from emergency bus. -Complete evaluation of SEP concerns: (a) Evaluate effect of missiles, pipe break, safety class & seismic req'ts (b) As part of main steam line break analysis, evaluate the need for capability to automatically terminate flow to a depressurized SG and feed the intact SG.

Table III-3 (W)

	Generic Recommendations													Plant Specific Recommendations	
	GS-1:Tech Spec LCO Train Outage Time Limit	GS-2:Tech Spec-Single Flow Path Manual Valves	GS-3:AFWS Flow Throttling-Water Hammer	GS-4:Emergency Procedure-Backup Water Supplies	GS-5:Emergency Procedure-AC Power Blackout	GS-6:Flow Path Verification	GS-7:Non-Safety Grade AFWS Automatic Start Signals	GS-8:Automatic Actuation of AFWS	GL-1:Automatic Actuation of AFWS	GL-2:Single Flow Path-Manual Valves	GL-3:Eliminate AC Power Dependency of One Train	GL-4:Multiple Pump Damage Protection-Natural Phenomena	GL-5:Non-Safety Grade AFWS Automatic Start Signals		
Plant	Short Term								Long Term					Short Term	Long Term
Zion 1&2 2-elect pumps* 1-turbine pump* *per unit Automatic Initiation	X	X	X	X	X	X	X			X	X	X	X	-Add Technical Specification LCO when Condensate Storage Tank level is below 170,000 gallons.	-Evaluate the need to qualify valves, valve operators and instrumentation for environmental conditions resulting from high energy line break. -Evaluate a break in the common headers from motor or turbine driven AFW pump to determine necessary design changes or procedure to preclude loss of all AFW flow to all steam generators.

REFERENCES

1. Standard Review Plan (NUREG-75/087), Chapter 10.4.9, and Branch Technical Position ASB 10-1.

2. Reactor Safety Study (WASH-1400), October 1975.

3. Memorandum, S. Levine to H. Lewis, dated June 27, 1978.

4. "Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission" (the "Lewis Committee Report"), NUREG/CR-0400.

5. Commission Policy on Risk Assessment.

6. Identification of Unresolved Safety Issues Relating to Nuclear Power Plants, NUREG-0510).

8. Paper presented at American Nuclear Society by S. Asselin, J. Hickman, et al, 1978.

9. IE Bulletin 79-06A.

10. IE Bulletin 79-06B.

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APPENDIX IV

CONTROL SYSTEMS

1. GENERAL FEATURES

The events at Three Mile Island Unit 2 (TMI-2) demonstrated that the cooling of the core by natural circulation can play an important role during the course of a small break loss-of-coolant accident (LOCA), particularly in the post-accident recovery phase of operation. Natural circulation can also play an important role during plant response to some anticipated transients. This section, therefore, addresses some of the control systems that potentially can cause such transients or influence natural circulation cooling. The effects of failures in these systems that might impede the development of adequate natural circulation cooling or lead to increased challenges to safety systems are also discussed.

Three basic requirements are necessary to achieve a stable natural circulation cooling mode in a pressurized water reactor with subcooled primary coolant in the steam generators and the eventual cooldown by natural circulation following some anticipated transients and postulated small breaks in the reactor coolant system:

(1) Establish a Sub-Cooled, Liquid Full Primary Coolant System
This requirement consists of refilling the reactor coolant system with liquid to eliminate, or prevent, the formation of pockets of steam and noncondensible gas at local or overall high points in the reactor coolant system, which could inhibit natural circulation. This requirement can be enhanced, for certain transients, by using high pressure safety injection or charging pumps, and by maintaining adequate control of the reactor coolant system pressure by using the pressurizer pressure and level control systems. When the reactor coolant system is full of liquid and subcooled, this requirement has been achieved.

(2) Establish a Heat Sink
This requirement can be met by providing steam generator secondary side coolant levels and temperatures that are sufficient to remove the stored and decay heat. With loss of offsite power, this requirement can be met by using the auxiliary feedwater system and the steam generator safety valves and the atmospheric dump valves.

(3) Establish Circulation
This requirement is met by establishing the density differences in the reactor coolant system to provide the driving force for natural circulation, using the auxiliary feedwater system, the steam generator safety valves, and the atmospheric dump valves.

2. PRESSURIZER PRESSURE CONTROL SYSTEM

The pressurizer pressure control system has the following components:

(1) Proportional and backup heaters establish pressure following certain transients and offset normal heat losses from the pressurizer.

(2) A pressurizer spray, which is supplied by the reactor coolant system cold leg, reduces pressure during some transients and prevents unnecessary opening of the pressurizer power-operated relief valves (PORVs). In some plants, the driving force for this spray is provided by the reactor coolant pumps. However, if the reactor coolant pumps are not operating, as would be the case during natural circulation, an auxiliary spray line from the makeup pumps in the chemical and volume control system (CVCS) can be used.

(3) Pressurizer PORVs limit transient pressure increases to values less than the setpoint of the pressurizer safety valves.

(4) Pressurizer safety valves prevent the maximum reactor coolant system pressure from exceeding 110 percent of the design pressure for anticipated operational occurrences.

In the natural circulation cooling mode, the pressurizer pressure could be controlled by means of the pressurizer heaters and auxiliary spray (if it is available), so that the entire reactor coolant system would remain in a subcooled condition during hot standby and subsequent cooldown and depressurization to cold shutdown conditions.

During normal operation, pressurizer level is controlled by a "feed and bleed" process. Charging water from the CVCS makeup pumps to the reactor coolant system is used to increase the level and to compensate for the reactor coolant pump seal leakoff losses. The level is decreased by letdown flow from the reactor coolant system to the CVCS. Pressurizer level during natural circulation flow conditions, such as at hot standby and during cooldown following some anticipated transients, is maintained by using the CVCS and the normal charging and letdown lines. The CVCS also serves to borate the reactor coolant system to (1) permit additional reduction in system temperature to that required for residual heat removal system operation, and, eventually, to the cold shutdown temperature and (2) maintain a subcritical condition.

The above-identified systems are described in this section of this report. More detailed, plant-specific information is presented in Appendices I and VI and Tables IV-1 and IV-2. Appendix V of this report discusses the safety systems which provide operator information. However, to improve understanding of this information, the following introductory comments are provided:

(1) Instrumentation or control system failures can possibly cause a transient. These failures may result in interactions requiring actions from one or more safety systems to limit the consequences to acceptable levels. The effects of these events are supposed to be scoped by the analyses which the applicant presents in Chapter 15 of the

TABLE IV-1. COMPARISON OF CONTROL SYSTEMS.

Plant Name	Power Sources: Control Instrumentation	Power Sources: Backup Heaters	Pressurizer Control: Set Point	Pressurizer Control: Gain/Repeat: Prop	Pressurizer Control: Gain/Repeat: Reset	Pressurizer Control: Gain/Repeat: Der	Heaters on 100% gain/psi	Spray	PORV #2 Lift Reseat	PORV #3 Lift Reseat
Beaver Valley 1	1E	1E	2235 psig	5	1.18	1 sec	2220 -3.333%	2260 2%	2335 2327	N/A
D. C. Cook 1&2	1E	Non 1E	2235 psig	5	15 min	1 sec	2220 -3.33%	2260 2%	2335 2315	2335 2315
Farley	1E	1E	2235 psig	4.02	5 min	N/A	NP	2260 2%	2335 2315	N/A
Ginna	1E	1E	NP	4	7 min	N/A	2220 -3.33%	2260 2%	2335 2185	N/A
Haddam Neck	Non-1E	Non 1E	N/A	N/A	N/A	N/A	2010 G -20%	2015 20%	2270 2250	N/A
H. B. Robinson	1E	1E	2235 psig (4 v = 2335)	1	3 min	10 sec	2220 -3.33%	2260 2%	4 V 3.96 V	N/A
Indian Point 2	1E	1E	2232 psig	1	3 min	10 sec	2220 -3.3%	2260 2%	2335 2320	N/A
Indian Point 3	NP	NP	NP	NP	NP	NP	NP NP	NP NP	NP NP	NP
Kewaunee	NP	1E	2235 psig 2235=30 5psi/ma	4	20 min	N/A	2200 NP	2260 2%	2315 2310	2335 2330
North Anna 1	1E	1E	2235 psig 2235=42.5 2315=82.5 2335=92.5	1	10 min	N/A	35% -6.67%	55% 4%	92.5% 82.5%	N/A
Point Beach 1	1E	1E	2235 psig	4	4 min	N/A	2220 -3.33%	2260 2%	2335 2315	N/A
Point Beach 2	1E	1E	2235 psig	4	7 min	N/A	2220 -3.33%	2265 2%	2335 2315	N/A
Prairie Island 1&2	1E	1E	2235 psig 2335=50 2315=46	5	20 min	N/A	2220 -3.33%	2270 2%	50 MA 46 MA	N/A

Note: "NP" means that the information was not available at report time.

TABLE IV-1 (Continued)

Plant Name	Power Sources		Pressurizer Control				Heaters on 100% gain/psi	Spray	PORV #2 Lift Reseat	PORV #3 Lift Reseat
	Control Instrumentation	Backup Heaters	Set Point	Gain/Repeat						
				Prop	Reset	Der				
Salem 1	NP	NP	NP	NP	NP	NP	NP	NP	2335 2315	N/A
San Onofre Unit 1	1E	1E	2085 (10/50)	30%	0.4 min	0.04 sec	15 MA -10%	30 MA 35 MA	47.5 MA 46.9 MA	N/A
Surry	1E	1E	2335 PSIG 2220=46.3 2250=53.8 2260-56.2 2310=68.8 2335=75	NP	NP	NP	53.8% -3.33%	56.2% 2%	NP	NP
Trojan	1E	1E	NP psig	NP	15	2	NP -3.33%	NP 2%	2335 2318	N/A
Turkey Point 3 & 4	1E	1E	2235 (92%=2335)	0.5	9.17 min	N/A	35% -20%/3%	55% 4%/%	92% 89%	NA
Yankee Rowe	Pressurizer = vital bus SG Level = vital bus SG pressure = Non 1E	Non-1E	N/A	N/A	N/A	N/A	1975 G -2%	2250 G 2%	N/A	
Zion 1&2	1E	1E	2235 psig	5	3 min	10 sec	NP 5.33	NP -3.2	2335 2325	N/A

Note: "NP" means that the information was not available at report time.

TABLE IV-2. COMPARISON OF REACTOR TRIP SETPOINTS.

	Pressurizer Scrams			Steam Generator Scrams		
Plant Name	High Pressure	High Level	Low Feedwater Flow #/Hr F Steam	Low-Low Level	Low Pressure	Low Reactor Coolant Flow
Beaver Valley 1	2385 psig	92%	NP	10%	500	90%
D. C. Cook 1	2385 psig	92%	NP	10%	600	90% (79,650 gpm/loop)
D. C. Cook 2	2385 psig	92%	NP	15%	600	90% (84,375 gpm/loop)
Farley	2385 psig	92%	1.5 x 106	15%	585	90%
Ginna	2377 psig	92%	8.0 x 107	30%	500	90%
Haddam Neck	2300 psig	86%	NP	NP	NP	90%
H. B. Robinson	2376 psig	91%	6.4 x 105	15%	NP	91%
Indian Point 2	2385 psig	92%	1.15 x 106	5%	600	90% (120,000 gpm/loop)
Indian Point 3	2365 psig	90%	30%	15%	600	90%
Kewaunee	2385 psig	90%	8.9 x 105	13%	500	90% (89,000 gpm/loop)
North Anna 1	NP	NP	NP	5%	600	NP
Point Beach 1 & 2	2365 psig	90%	8 x 105	10%	500	93%
Prairie Island 1 & 2	2385 psig	86.5%	8.9 x 105	13%	500	90%
Salem 1	2385 psig	92%	1.42 x 106	5%	500	90%
San Onofre 1	2220 psig	27.3 ft	25%	None	None	85% (69,560 gpm/loop)
Surry 1	NP	NP	7.05 x 105	13%	525	90.5%
Surry 2	NP	NP	7.09 x 105	13%	525	92%
Trojan	2385 psig	92%	40%	5%	None	90% (90.270 gpm/loop)
Turkey Point 3&4	2385 psig	92%	6.4 x 105	15%	600	90%
Yankee Rowe	None	200 in.	None	13 in.	None	80%
Zion 1&2	NP psig	NP	7 x 105	10%	600	NP

Note: "NP" means that the information was not available at report time.

Safety Analysis Reports. However, recent events have raised a concern about the possible interaction between safety-grade systems and non-safety grade systems. The Lessons Learned Task Force has recommended that all licensees with operating reactors and applicants holding construction permits be required to evaluate the interaction of non-safety and safety-grade systems during normal operation, transients, and design basis accidents to assure that any interaction will not result in exceeding the acceptance criteria for any design basis event.

(2) The operator can change the course of an event by using the manual controls, based on his interpretation of the multiple alarms and instruments which are provided in a typical nuclear power station.

(3) Because of the long transport delays and large heat capacities which result from the large physical size of the fluid systems in a nuclear power plant, ample time is usually available for an operator to take corrective action and compensate for failures in nonsafety control systems. Typical corrective actions include the following:

   (a) Closing of block valves,

   (b) Placing control systems in manual control,

   (c) Selecting an alternate signal source, and

   (d) Selecting an alternate control strategy (e.g., changing control mode or plant operating state).

(4) The non-Class 1E control systems are not usually connected to onsite power sources. Therefore, automatic control of non-Class 1E systems should not be relied upon in the event of a loss of offsite power.

(5) The transient analyses which are usually performed are based on the assumption that a single system or module fails. However, a single power source may be used for most of the non-Class 1E instrumentation and controls. Thus the failure of a single bus could initiate multiple transients. The cumulative effects fo these transients are not analyzed, although Chapter 15 transients may bound these failures, but this needs to be confirmed. In addition to power supply problems, other control system and process interactions could lead to plant trip. Some control systems share sensors with each other, and with safety systems. Therefore, the plant control process system could provide a feedback path to the safety systems. Such interactions are being studies by Sandia Laboratories under a contract with the NRC. Some brief examples of such interactions that are based on the designs of non-Class IE systems are described in this report.

2.1 Pressurizer Level

The pressurizer level control system establishes, maintains, and restores pressurizer water level within specified limits, as a function of the average coolant temperature. Changes in level are caused by operational transients, which may cause the fluid to shrink or swell. Controlled level changes are produced by means of charging flow control (manual or automatic), as well as by manual selection of letdown orifices. Maintaining coolant level in the pressurizer within prescribed limits by actuating the charging and letdown system provides control of the reactor coolant water inventory.

The pressurizer level control system is a subsystem of the chemical and volume control system. During normal plant operation, the charging flow varies to produce the flow demanded by the pressurizer water level controller. The pressurizer water level is programmed as a function of average coolant temperature, with the highest average temperature (auctioneered) being used. The pressurizer water level setpoint decreases as the load is reduced from full load (25% level at 0% power and 50% level at 110% power). The programmed level is designed to match, as nearly as possible, the level changes resulting from the coolant temperature changes.

To control pressurizer water level during startup and shutdown operations, the charging flow is manually regulated from the main control room.

A block diagram of a typical pressurizer water level control system is shown in Figure IV-1.

The auctioneered average temperature (T_{avg}) and the pressurizer level signals are derived from the Class 1E reactor protection system. The safety signals are isolated from the control signals in accordance with the requirements of IEEE Standard 279. An individual input to the temperature auctioneer may be defeated by a console-mounted bypass switch, but a failure in the auctioneer will cause a failure in both the pressurizer level control and steam dump control systems. By means of a switch in the control room an operator may select one of two pressurizer level signals for pressurizer level control. Thus, the operator has the capability to reconfigure parts of this control system to compensate for some module failures.

Maintaining an adequate pressurizer level is not always sufficient to assure that proper reactor coolant inventory is being maintained. As a result of the TMI-2 accident, the need to obtain a more accurate measure of reactor coolant inventory has been recognized. The two systems for making such measurements which are most often mentioned are level measurement and inventory control. The requirements specified by NUREG-0578 regarding instrumentation to detect inadequate core cooling address this concern.

2.2 Pressurizer Pressure

The pressurizer pressure control system maintains or restores the pressurizer pressure to the desired pressure ±35 psi (which is well within reactor trip and relief and safety valve actuation setpoint limits), following normal operational transients that induce pressure changes. This is accomplished by control (manual or automatic) of heaters and spray in the pressurizer. The pressurizer control system also provides steam relief by controlling the PORVs, which are discussed in Appendix VI of this report. In addition, the level system will energize the heaters on higher level and de-energize them on low level.

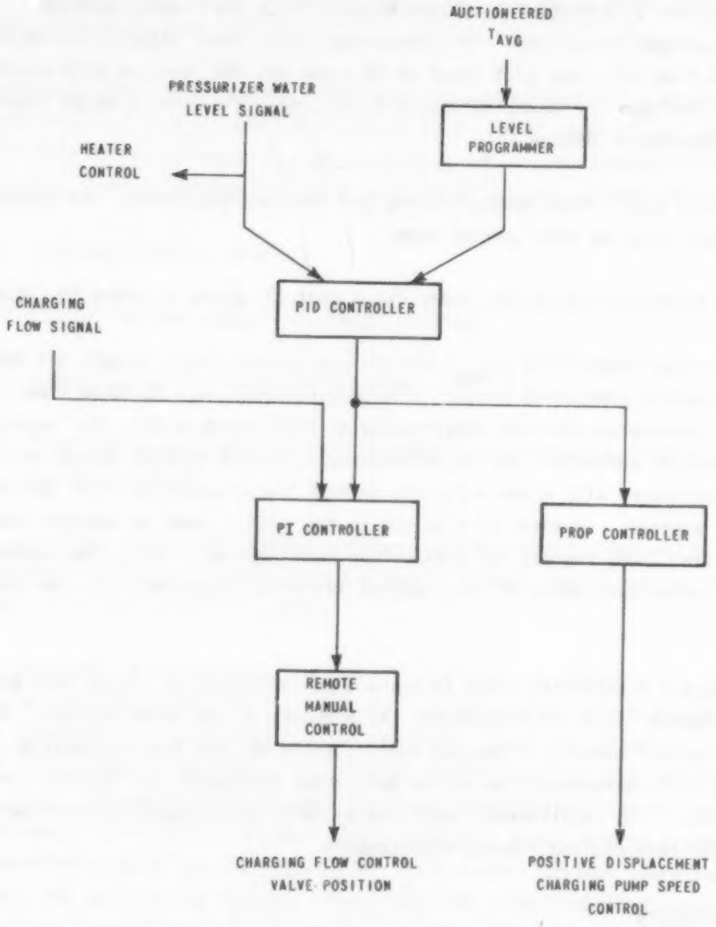
AUCTIONEERED
T_{AVG}
PRESSURIZER WATER
LEVEL SIGNAL
LEVEL
PROGRAMMER
HEATER
CONTROL
CHARGING
FLOW SIGNAL
PID CONTROLLER
PI CONTROLLER
PROP CONTROLLER
REMOTE
MANUAL
CONTROL
CHARGING FLOW CONTROL
VALVE POSITION
POSITIVE DISPLACEMENT
CHARGING PUMP SPEED
CONTROL

Figure IV-1. Block Diagram of Pressurizer Level Control System.

The reactor coolant system pressure is normally controlled by using either the pressurizer heaters (in the water region), or the pressurizer spray (in the steam region), plus steam relief for large transients. The electrical immersion heaters are located near the bottom of the pressurizer. A portion of the heater group is proportionally controlled to correct small pressure variations. These variations are due to heat losses, including heat losses due to a small continuous spray. The remaining (backup) heaters are turned on when the pressurizer pressure controller signal demands approximately 100 percent proportional heater power.

The spray nozzles are located on the top of the pressurizer. Spray is initiated when the pressure controller spray demand signal is above a given setpoint. The spray rate increases proportionally with increasing spray demand signal, until it reaches a maximum value.

Steam condensed by the spray reduces the pressurizer pressure. A small continuous spray is normally maintained to reduce thermal stresses and thermal shock, and to help maintain uniform water chemistry and temperature in the pressurizer.

The spray source is the discharge from one of the reactor coolant pumps. However, some plants also have an "auxiliary spray system" which is driven from the discharge of the charging pumps. The normal power source for the pressurizer heaters and the auxiliary sprays is a non-Class 1E bus. However, several plants (see Table IV-1) have the capability of powering the backup heaters from an onsite source, when diesel generator loading conditions permit a manual transfer of this load.

In W-designed operating plants, there are two separate controls for the PORVs. The first valve is controlled by a bistable in a pressure measurement channel. When a given pressure is exceeded, the valve opens and will not reseat until the pressure has been reduced below the bistable hysteresis point. The second (and third in a plant with 100% load rejection capability) valve is controlled by the pressure control sytem. Often (see Table IV-1) this system has reset and derivative gains have resulted in an output that anticipates the actual pressure. This "corrected" pressure can result in a substantial reduction in actual PORV lift point for transients which have a high rate of pressure increase, or which require sustained operation of the spray valves. A block diagram of the pressurizer pressure control system is shown in Figure IV-2. An interlock from a second pressure channel is provided to prevent a single failure in the control system from causing a spurious lifting of the PORVs.

The control room operator can select an alternate pressurizer pressure sensor (Class 1E) by the use of a control console switch.

The failure of a PORV to reclose following an overpressure transient was an important factor during the TMI-2 accident. The PORVs can be operated either manually or automatically. The control circuits for these valves currently are not single failure-proof. That is, a single failure in the control circuit can result in a small LOCA, which could involve all PORVs opening concurrently. Block valves are provided upstream of the relief valves to isolate such failures. In the event of the loss of offsite power (which in all probability will result in a feedwater transient), the operator would not have the capability for controlling

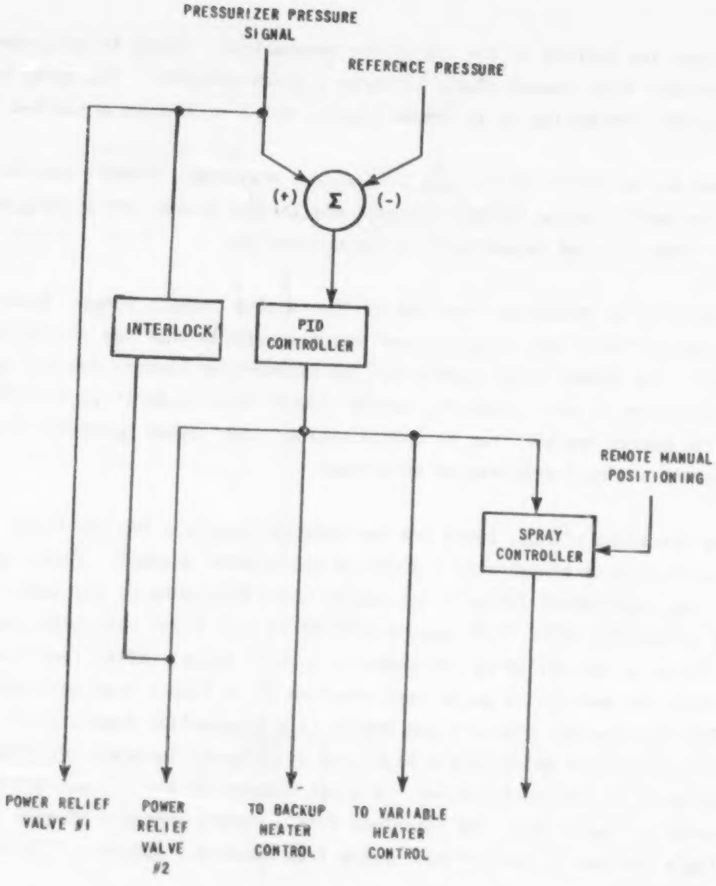
PRESSURIZER PRESSURE SIGNAL
REFERENCE PRESSURE
(+)
Σ
(-)
INTERLOCK
PID CONTROLLER
REMOTE MANUAL POSITIONING
SPRAY CONTROLLER
POWER RELIEF VALVE #1
POWER RELIEF VALVE #2
TO BACKUP HEATER CONTROL
TO VARIABLE HEATER CONTROL

Figure IV-2. Block Diagram of Pressurizer Pressure Control System.

the operation of the PORVs or for isolating a stuck open PORV, if both the PORVs or the associated block valves, respectively, were not powered from the emergency buses. NUREG-0578 has specified a number of requirements regarding emergency power supply requirements for the pressurizer heaters, pressurizer PORVs and block valves, pressurizer level indicators, and relief and safety valve performance testing that address these concerns.

Table IV-1 indicates the plants that, at the time this report was written, have their instrumentation and controls (pressurizer pressure and level, steam generator pressure and level) on Class 1E power sources. The PORVs are dc solenoid-operated. The instrumentation power is inverted ac power, supplied from the same division.

2.3 Steam Generator Water Level

Westinghouse (W)-designed plants have two different systems for maintaining steam generator water level. These systems may be classified as the normal and auxiliary feedwater systems. In most W-designed plants, the auxiliary system is a Class 1E system, with its own controls. The normal non-Class 1E main feedwater system is discussed here. The Class 1E auxiliary feedwater system is discussed in Appendix III.

The steam generator water level control system establishes and maintains the steam generator water level to within predetermined physical limits during normal operating transients. This system also restores the steam generator water level to predetermined limits after a reactor trip and regulates the feedwater flow rate such that, under operational transients, the heat sink for the reactor coolant system is maintained.

Each steam generator is equipped with a three-element feedwater flow controller that maintains a programmed water level, which is a function of reactor power level. The three-element feedwater controller regulates the feedwater valve by continuously comparing the feedwater flow signal, the water level signal, the programmed level and the pressure-compensated steam flow signal. In addition, the feedwater pump speed is varied to maintain a programmed pressure differential between the steam header and the feed pump discharge header. The speed controller continuously compares the actual pressure differential (ΔP) with a programmed ΔP_{ref}, which is a linear function of steam flow. Continued delivery of feedwater to the steam generators is required as a sink for the heat stored and generated in the reactor following a reactor trip and a turbine trip. An override signal closes the feedwater valves when the average coolant temperature is below a given temperature, and the reactor has tripped. Manual override of the feedwater control system is available at all times.

A block diagram of a typical steam generator water level control system is shown in Figures IV-3 and IV-4. For each three-element control system, the operator can select the following alternate Class 1E sensors by the use of control console-mounted switches:

(1) steam flow, or

(2) feedwater flow.

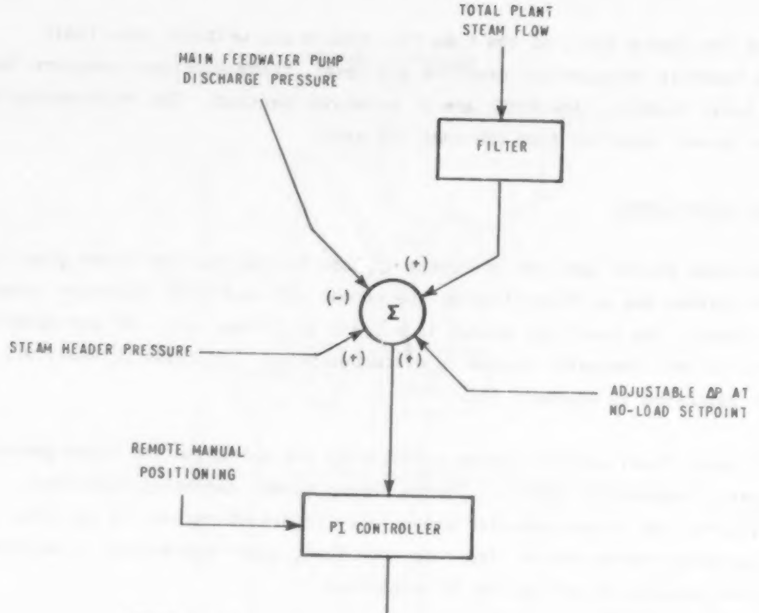
TOTAL PLANT STEAM FLOW
MAIN FEEDWATER PUMP DISCHARGE PRESSURE
FILTER
(+)
(-)
Σ
STEAM HEADER PRESSURE
(+)
(+)
ADJUSTABLE ΔP AT NO-LOAD SETPOINT
REMOTE MANUAL POSITIONING
PI CONTROLLER
REMOTE MANUAL POSITIONING
PROP CONTROLLER
(TYPICAL-EACH PUMP HAS ITS OWN PROP CONTROLLER)
MAIN FEEDWATER PUMP SPEED

Figure IV-3. Block Diagram of Main Feedwater Pump Speed Control System.

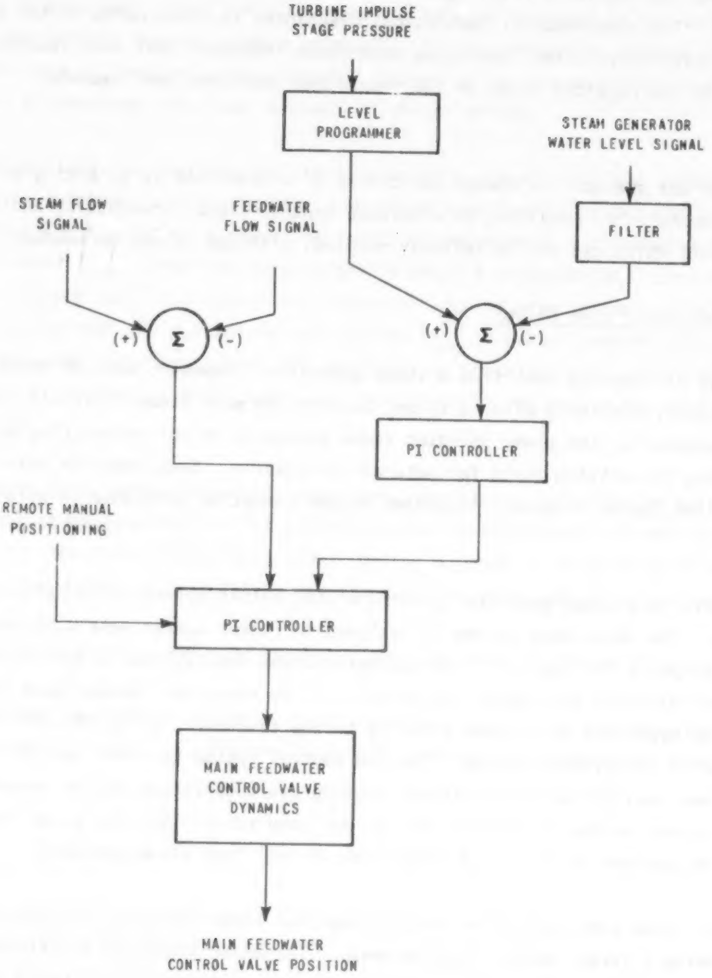
TURBINE IMPULSE
STAGE PRESSURE
LEVEL
PROGRAMMER
STEAM GENERATOR
WATER LEVEL SIGNAL
STEAM FLOW
SIGNAL
FEEDWATER
FLOW SIGNAL
FILTER
(+)
Σ
(-)
(+)
Σ
(-)
PI CONTROLLER
REMOTE MANUAL
POSITIONING
PI CONTROLLER
MAIN FEEDWATER
CONTROL VALVE
DYNAMICS
MAIN FEEDWATER
CONTROL VALVE POSITION

Figure IV-4. Block Diagram of Steam Generator Water Level Control System.

The other sensors are not classified as Class 1E. The alternative of selecting alternate sensors by the use of switches is not provided.

For purposes of simplicity and conservatism with regard to secondary water inventory, a proportional integral derivative a (PID) type controller is not provided in most plant transient analyses. The occurrence of theoretical resonances in other parts of the system is neglected. As previously stated, operating experience indicates that such resonances do not occur because of the physical sizes of the plants and resultant low frequency characteristics.

The opportunity for the operator to change the course of a transient by selecting an alternate signal source or by providing an alternate control signal presents an additional combination of events which can not be reliably modeled, although it can be bounded.

2.4 Steam Generator Shell Side Relief

There are many ways of removing heat from a steam generator. However, most of these methods will not be immediately available after a scram, because the main steam lines will usually be isolated. Consequently, the power-operated steam generator relief valves play a significant role in establishing the driving force for natural circulation. Even when the main steam line isolation valves can be reopened, condenser vacuum cannot be maintaned if offsite power is lost.

In W-designed plants, the steam generator power-operated relief valves are a part of the steam dump system. The steam dump system is designed to reject approximately 50 percent of net load without tripping the reactor. The automatic steam dump system is able to accommodate this abnormal load rejection and reduce the effects of the transient imposed upon the reactor coolant system. By bypassing main steam directly to the condenser, an artificial load is thereby maintained on the primary system. The rod control system can then reduce the reactor temperature to a new equilibrium value without causing overtemperature and/or overpressure conditions. For plants having 50 percent loss of net load capability, the steam dump steam flow capacity is 40 percent of full load steam flow at full load steam pressure.

The load rejection steam dump controller circuit prevents large increases in reactor coolant temperature following a large, sudden load decrease. The error signal is a difference between the lead/lag-compensated, auctioneered T_{avg} and the reference T_{avg}, based on turbine impulse chamber pressure.

The T_{avg} signal is the same as that used in the reactor coolant system. The lead/lag compensation for the T_{avg} signal is to compensate for lags in the plant thermal response and in valve positioning. Following a sudden load decrease, T_{ref} is immediately decreased and T_{avg} tends to increase, thus generating an immediate demand signal for steam dump. Since control rods are usually available in this situation, steam dump terminates as the error comes within the maneuvering capability of the control rods.

If the error signal exceeds a predetermined amount, a demand signal will actuate the steam dump to maintain the reactor coolant system temperature within control range, until a new equilibrium condition is reached.

To prevent actuation of steam dump on small load perturbations, an independent load rejection sensing circuit is provided. This circuit senses the rate of decrease in the turbine load, as detected by the turbine impulse chamber pressure. It is provided to unblock the dump valves, when the rate of load rejection exceeds a preset value corresponding to a 10% step load decrease or a sustained ramp load decrease of 5% per minute.

Following a turbine trip, as monitored by the turbine trip signal, the load rejection steam dump controller is defeated, and the turbine trip steam dump controller becomes active. The demand signal is the error signal between the lead/lag-compensated, auctioneered T_{avg} and the no load reference T_{avg}. When the error signal exceeds a predetermined setpoint, the dump valves are tripped open in a prescribed sequence. As the error signal reduces in magnitude, indicating that the reactor coolant system T_{avg} is being reduced toward the reference no-load value, the dump valves are modulated by the plant trip controller to regulate the rate of removal of decay heat, and thus gradually establish the equilibrium hot shutdown condition.

Following a turbine trip, only sufficient steam dump capacity is necessary to maintain steam pressure below the steam generator relief valve setpoint (approximately 40 percent capacity to the condenser). The error signal determines whether a group is to be tripped open or modulated open. The valves are modulated when the error is below the trip-open setpoints.

A block diagram of a typical steam dump control system is shown in Figure IV-5.

As previously stated, the design of the controls includes a provision for manual control. In this system, manual control is used for long-term cooldown to remove decay heat.

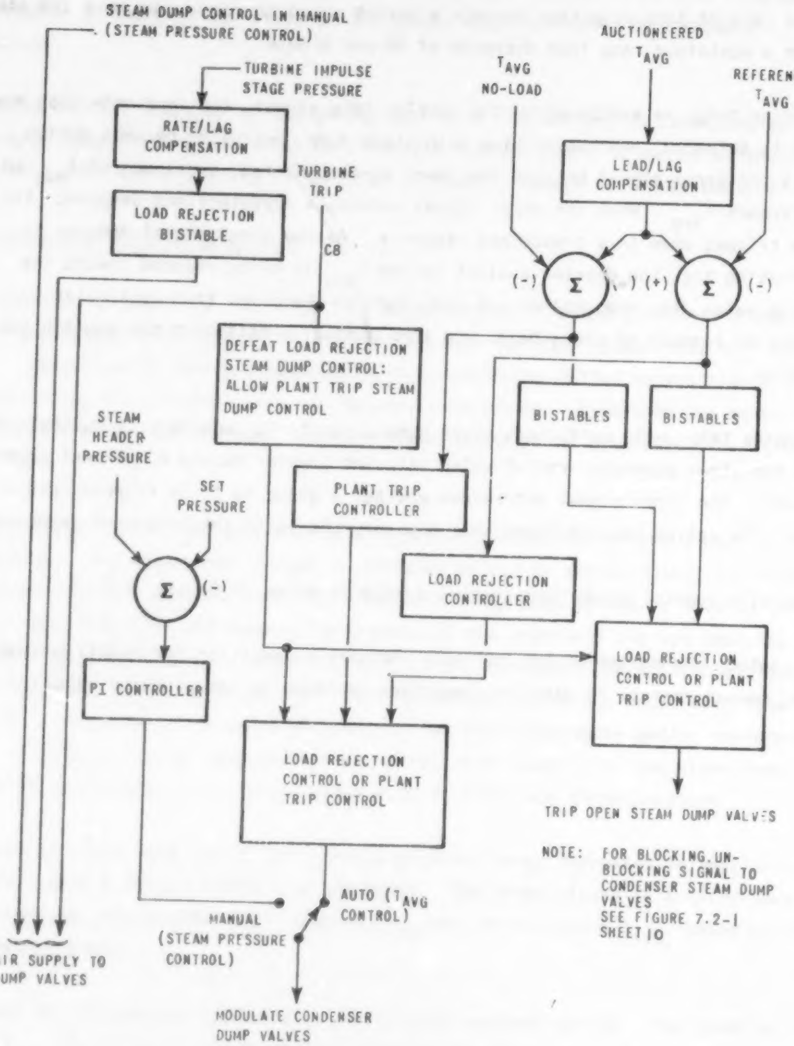
STEAM DUMP CONTROL IN MANUAL
(STEAM PRESSURE CONTROL)
TURBINE IMPULSE
STAGE PRESSURE
RATE/LAG
COMPENSATION
TURBINE
TRIP
LOAD REJECTION
BISTABLE
CB
DEFEAT LOAD REJECTION
STEAM DUMP CONTROL:
ALLOW PLANT TRIP STEAM
DUMP CONTROL
STEAM
HEADER
PRESSURE
SET
PRESSURE
Σ
(-)
PI CONTROLLER
PLANT TRIP
CONTROLLER
LOAD REJECTION
CONTROLLER
LOAD REJECTION
CONTROL OR PLANT
TRIP CONTROL
AUCTIONEERED
T_{AVG}
T_{AVG}
NO-LOAD
REFERENCE
T_{AVG}
LEAD/LAG
COMPENSATION
(-)
Σ
(+)
(+)
Σ
(-)
BISTABLES
BISTABLES
LOAD REJECTION
CONTROL OR PLANT
TRIP CONTROL
TRIP OPEN STEAM DUMP VALVES
NOTE: FOR BLOCKING, UN-
BLOCKING SIGNAL TO
CONDENSER STEAM DUMP
VALVES
SEE FIGURE 7.2-1
SHEET 10
AUTO (T_{AVG}
CONTROL)
MANUAL
(STEAM PRESSURE
CONTROL)
AIR SUPPLY TO
DUMP VALVES
MODULATE CONDENSER
DUMP VALVES

Figure IV-5. Block Diagram of Steam Dump Control System.

APPENDIX V

SAFETY SYSTEMS

1. GENERAL FEATURES

The previous discussion of non-Class 1E control systems in Appendix IV described how failures in such systems could challenge the reactor protection system (RPS) in Westinghouse (W)-designed reactors. It was shown that, in some cases, the non-safety-grade control systems and the RPS measure the same parameters, use the same sensors, and operate on the same variables. The sections that follow describe the generic aspects of the RPS. More specific information is presented in Tables V-1 and V-2.

The RPS used in W-designed reactors consists of two major subsystems: (1) the reactor trip system (RTS), and (2) the engineered safety features actuation system (ESFAS), each of which is described in the sections that follow.

2. REACTOR TRIP SYSTEM (RTS)

The RTS used in W designed reactors contains two trains, A and B, each of which is capable of opening a separate and independent reactor trip breaker to cause a reactor scram. Each train of the RTS includes a prescribed number of redundant channels for monitoring various plant parameters. The logic associated with each train is such that, when the prescribed number of parameters (the number of which are indicated in Table V-2) are outside the safe operating range, the reactor will be automatically tripped.

The RTS automatically keeps the reactor operating within a safe region by shutting down the reactor whenever the limits of the region are exceeded. The safe operating region is defined by several considerations, such as mechanical/hydraulic limitations on equipment and heat transfer phenomena. Therefore, the RTS surveys process variables related to equipment limitations, such as pressure, pressurizer water level, and also variables which directly affect the heat transfer capability of the reactor (e.g., flow and reactor coolant temperatures). Other parameters utilized in the RTS are calculated from various process variables. Whenever a direct process or calculated variable exceeds a setpoint, the reactor will be shut down to preclude fuel damage or the release of radioactive effluents to the containment and the environment.

The RTS also initiates a turbine trip signal upon reactor trip. This action prevents a positive reactivity insertion that could otherwise result from excessive reactor system cooldown, and avoids unnecessary actuation of the ESFAS that could result due to shrinkage of the primary fluid. The RTS design also provides for a manual trip of the reactor by operator action.

TABLE V-1
PROTECTION SYSTEM INTERLOCKS

Designation	Derivation	Function
	I POWER ESCALATION PERMISSIVES	
P-6	Presence of P-6: 1/2 neutron flux (intermediate range) above setpoint	Allows manual block of source range reactor trip.
	Absence of P-6: 2/2 neutron flux (intermediate range) below setpoint	Defeats the block of source range reactor trip.
P-10	Present of P-10: 2/4 neutron flux (power range) above setpoint	Allows manual block of power range (low setpoint) reactor trip.
		Allows manual block of intermediate range reactor trip and intermediate range rod stops (C-1).
		Blocks source range reactor trip (back-up for P-6).
	Absence of P-10: 3/4 neutron flux (power range) below setpoint trip.	Defeats the block of power range (low setpoint) reactor
		Defeats the block of intermediate range reactor trip and intermediate range rod stops (C-1).
		Input to P-7.
	II BLOCKS OF REACTOR TRIPS	
P-7	Absence of P-7: 3/4 neutron flux (power range) below setpoint (from P-10)	Blocks reactor trip on: Low reactor coolant flow in more than one loop, undervoltage.
	2/2 turbine impulse and chamber pressure below setpoint (from P-13)	Reactor coolant pump breaker open in more than one loop, undervoltage, underfrequency, turbine trip, pressurizer low pressure, and pressurizer high level.
P-8	Absence of P-8: 3/4 neutron flux (power range) below setpoint	Blocks reactor trip on low reactor coolant flow or reactor coolant pump breaker open in a single loop.
P-13	2/2 turbine impulse chamber pressure below setpoint	Input to P-7..

TABLE V-2

LIST OF REACTOR TRIPS

Reactor Trip	Coincidence Logic	Interlocks	Comments
1. High neutron flux (Power Range)	2/4	Manual block of low setting permitted by P-10	High and low setting; manual block and automatic reset of low setting by P-10.
2. Intermediate range neutron flux	1/2	Manual block permitted by P-10	Manual block and automatic reset.
3. Source range neutron flux	1/2	Manual block permitted by P-6, interlocked with P-10	Manual block and automatic reset. Automatic block above P-10.
4. Power range high positive neutron flux rate	2/4	No interlocks	
5. Power range high negative neutron flux rate	2/4	No interlocks	
6. Overtemperature differential temperature ΔT	2/4	No interlocks	
7. Overpower ΔT	2/4	No interlocks	
8. Pressurizer low pressure	1/3	Interlocked with P-7 and lower pressurizer level	Blocked below P-7.
9. Pressurizer high pressure	2/4	No interlocks	

TABLE V-2 (Continued)

Reactor Trip	Coincidence Logic	Interlocks	Comments
10. Pressurizer high water level	2/3	Interlocked with P-7	Blocked below P-7.
11. Low reactor coolant flow	2/3 in any loop	Interlocked with P-7 and P-8	Low flow in one loop will cause a reactor trip when above P-8 and a low flow in two loops will cause a reactor trip when above P-7. Blocked below P-7.
12. Reactor coolant pump bus undervoltage	2/4	Interlocked with P-7	Low voltage on all buses permitted below P-7.
13. Reactor coolant pump bus underfrequency	2/4	Interlocked with P-7	Underfrequency on two buses will trip all reactor coolant pump breakers and cause reactor trip; reactor trip blocked below P-7.
14. Low feedwater flow	1/2 in any loop*	No interlocks	
15. Low-low steam generator water level	2/3 in any loop	No interlocks	
16. Safety injection signal	Coincident with actuation of safety injection	No interlocks	(See Section 7.3 for Engineered Safety Features actuation conditions.)

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*1/2 steam/feedwater flow mismatch in coincidence with 1/2 low steam generator water level.

TABLE V-2 (Continued)

Reactor Trip	Coincidence Logic	Interlocks	Comments
17. Turbine-generator trip**			
a. Low auto stop oil pressure	2/3	Interlocked with P-7	Blocked below P-7.
b. Turbine stop valve close	4/4	Interlocked with P-7	Blocked below P-7.
18. Manual	1/2	No interlocks	

**Reactor trip on turbine trip is anticipatory in that no credit is taken for it in accident analyses.

The specific reactor trips generated by the RTS are described below.

2.1 Power Range High Neutron Flux Trip

The power range high neutron flux trip circuit trips the reactor whenever two out of the four power range channels exceed the trip setpoint.

2.2 Intermediate Range High Neutron Flux Trip

The intermediate range high neutron flux trip circuit trips the reactor whenever one out of the two intermediate range channels exceeds the trip setpoint.

2.3 Source Range High Neutron Flux Trip

The source range high neutron flux trip circuit trips the reactor whenever one out of the two source range channels exceeds the trip setpoint.

2.4 Power Range High Positive Neutron Flux Rate Trip

This circuit trips the reactor whenever a sudden abnormal increase in neutron flux occurs in two out of the four power range channels.

2.5 Power Range High Negative Neutron Flux Rate Trip

This circuit trips the reactor upon a rapid decrease in nuclear power in two out of the four power range channels.

2.6 Core Thermal Overpower Trips

W-designed reactors are provided with the following two core thermal overpower trips:

(1) Overtemperature Differential Temperature (ΔT) Trip.

(2) Overpower ΔT Trip.
(This trip protects the core against a low departure from nucleate boiling ratio and trips the reactor whenever two out of four core thermal channels exceed their trip setpoint.)

2.7 Reactor Coolant System Pressurizer Pressure and Water Level Trips

2.7.1 Pressurizer Low Pressure Trip

This trip assures that a departure from nucleate boiling condition would not be reached if the primary pressure falls below preset limits.

2.7.2 Pressurizer High Pressure Trip

This trip protects the reactor coolant system against system overpressure. A trip is produced whenever two out of the four pressurizer pressure signals exceed the preset limit.

2.7.3 Pressurizer High Water Level Trip

This trip is a backup to the high pressurizer pressure trip and serves to prevent water relief through the pressurizer safety valves.

2.8 Reactor Coolant System Low Flow Trips

These trips protect the core from a departure from nucleate boiling condition in the event of loss of coolant flow. The means of sensing the loss of coolant flow are as follows:

2.8.1 Low Reactor Coolant Flow Trip

Reactor coolant flow sensors (three in each loop) provide a low flow indication for that loop. The indication in one loop (two out of three logic) will initiate a trip of the reactor if the power level is above that acceptable for three-pump operation. Similarly, a low flow indication in two loops (two out of three logic) will initiate a trip if the power level is above that acceptable for two-pump operation.

2.8.2 Reactor Coolant Pump Undervoltage Trip

Undervoltage relays are used to sense low voltage to the reactor coolant pump motors. The reactor will be tripped whenever a prescribed low voltage condition exists on more than one pump motor. However, this low voltage condition must exist for a sufficient time to preclude spurious trips due to short-term voltage fluctuations.

2.8.3 Reactor Coolant Pump Underfrequency Trip

The setpoint of the underfrequency relays is adjustable between 54 and 59 Hertz (Hz).

There is one underfrequency sensing relay for each reactor coolant pump motor. Signals from any two such relays will trip the reactor if the power level is above preestablished values. However, these trips are time delayed up to approximately 0.1 second to prevent spurious trips caused by short-germ frequency perturbations.

In newer plants, the only inputs associated with the reactor coolant pump trip are from the undervoltage and underfrequency sensors which are located on the load side of the reactor coolant pump breaker. These sensors are housed within a seismic Category I structure, and are designed in accordance with the requirements of IEEE Standard 279-1971.

In older plants, an auxiliary set of contacts is provided on the reactor coolant pump breakers to provide an anticipatory trip.

2.9 Steam Generator Trips

2.9.1 Low Feedwater Flow Trip

This trip protects the reactor from a sudden loss of heat sink. The trip is actuated by a steam flow/feedwater flow mismatch in coincidence with low water in any steam generator. The coincidence logic and interlocks for this trip are shown in Table V-2.

2.9.2 Low-Low Steam Generator Water Level Trip

This trip protects the reactor from loss of heat sink in the event of a sustained steam flow/feedwater flow mismatch of sufficient magnitude to cause a low feedwater flow reactor trip. This trip is actuated on two out of three low-low water level signals occurring in any steam generator.

2.10 Reactor Trip on a Turbine Trip

Reactor trip on a turbine trip is the only anticipatory trip included in the reactor protection system. This trip is anticipatory in that it causes a reactor trip before any of the trips described in Chapter 15 of the safety analysis report would reach their trip setting. This trip meets the IEEE Standard 279-1971 requirements pertaining to separation, redundancy, and testability. However, because the sensors are located in the turbine building, a non-seismic Category I building, the trip does not meet the seismic design requirements of that standard.

This trip is actuated upon low autostop oil pressure or upon the closure of the turbine steam stop valves, in accordance with the logic indicated in Table V-2, whenever the reactor is operating above P-7.

2.11 Safety Injection Signal Actuation Trip

This trip protects the core upon a loss-of-coolant accident or a main steam line break. The trip is automatically initiated whenever a safety injection system is actuated.

3. ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS)

The ESFAS consists of (1) an analog portion which monitors various plant parameters, and (2) a digital portion, which actuates engineered safety features (ESF) equipment upon certain prescribed conditions. The analog portion includes three or four redundant channels for each parameter monitored. The digital portion consits of two redundant trains, each of which receives input from the analog channels and actuates the appropriate ESF equipment if a sufficient number of parameters exceed the prescribed safety limits.

The ESFAS uses selected plant parameters to determine whether there is a primary or a secondary system pipe rupture.

In general, the ESFAS uses the same parameters and sensors used by the RTS. Additional parameters used by the ESFAS, however, are steam line pressure, containment pressure, steam line differential pressure, and steam flow. In addition, plants having loop stop valves use the loop stop valve position as an input to the ESFAS in order to select the appropriate combination of steam line pressure sensors.

The specific functions which rely on the ESFAS for initiation are:

(1) Reactor trip, if the trip has not been previously initiated by the RTS.

(2) Opening of the cold leg injection isolation valves. These valves are opened so that highly borated water may be injected into the reactor coolant system by the safety injection pumps.

(3) Starting of the charging pumps, safety injection pumps, residual heat removal pumps and opening of associated valves to provide emergency cooling water to the cold legs of the reactor coolant system.

(4) Closure of the upper head injection accumulator isolation valve on low water level signal from the accumulator (if upper head injection is provided).

(5) Starting essential fan cooling water and component cooling water pumps and isolating nonessential systems.

(6) Starting electric motor- and steam turbine-driven auxiliary feedwater pumps.

(7) Initiation of Phase A containment isolation to prevent fission product release.

(8) Steam line isolation to prevent the continuous, uncontrolled blowdown of more than one steam generator, and thereby uncontrolled reactor coolant system cooldown.

(9) Main feedwater line isolation, as required, to prevent or mitigate the effect of excessive cooldown.

(10) Starting of emergency diesels to assure backup supply of power to emergency systems.

(11) Isolating of the control room intake ducts to meet control room occupancy requirements.

(12) Actuating the emergency gas treatment system.

(13) Isolating the containment ventilation system.

(14) Actuating containment spray, upon a high containment condition, to perform the following functions:

(a) Reduce containment pressure and temperature following a loss-of-coolant accident (LOCA) or steam line break inside of containment.

(b) Initiate Phase B containment isolation, which isolates the containment following a LOCA, or a steam or feedwater line break within containment to limit radioactive releases. (Phase B isolation together with Phase A isolation results in isolation of all but safety injection and spray lines penetrating the containment.)

(15) Automatic switchover of the residual heat removal system pumps from the injection to the recirculation mode in newer plants.

APPENDIX VI

PRESSURIZER POWER-OPERATED RELIEF VALVES AND SAFETY VALVES

1. POWER-OPERATED RELIEF VALVE CONSIDERATIONS

The failure of a power-operated relief valve (PORV) to reclose following the overpressure transient was a key factor during the Three Mile Island Unit 2 (TMI-2) accident. This appendix discusses the design of this component and previous operating experiences related thereto in operating plants with Westinghouse (W)-designed reactors.

1.1 Design

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereinafter referred to as "the Code") requires that the reactor coolant system be protected from transient overpressure conditions. This protection is accomplished by several means, including reactor trip, operation of the Code required safety valves, and operation of the PORVs.

All W-designed operating reactors are equipped with one to three PORVs. Most plants have two, only Yankee Rowe has one, while two plants have three (see Table VI-1). The PORVs are designed to prevent the lifting of the pressurizer Code safety valves and to allow the reactor to remain on the line for load rejection transients. The PORVs are not considered part of the plant's safety systems and therefore no credit is taken for them in the safety systems. However, on most plants this valve(s) is used to prevent overpressurization of the reactor vessel during operation at low temperature, hence Technical Specifications will exist for this purpose.

The PORV used on W-designed operating plants (except for Yankee Rowe, which has an electromatic relief valve like that used on Combustion Engineering (CE) - and Babcock and Wilcox (B&W)-designed plants) is a spring-loaded valve with an air actuating opening. This will overcome the spring force on the valve stem and open the valve. Closure of the valve is initiated by venting air off the control diaphragm causing the spring force to positively seat the valve closed. The valve will close on loss of air. Plant-specific PORV and safety valve data are provided in Table VI-1.

TABLE VI-1

CODE SAFETY-RELIEF VALVES & POWER-OPERATED
RELIEF VALVES ON PRESSURIZER

	Plant Name						
	Farley 1	H. B. Robinson	Zion 1&2	Haddam Neck	Indian Point 2&3	Beaver Valley 1	Turkey Point 3&4
Code Safety Relief Valves							
MFG	Crosby	Crosby	Crosby	Crosby	Crosby	Target Rock	Turkey Point 3&4
Number	3	3	3	3	3	3	3
Type	E Spring loaded	Spring loaded	Spring operated	flanged spring loaded	Spring loaded	Spring loaded	Spring loaded
Model No.	HB-BP-86	HB-86-BP	RV-58-MSB	3K26H15-86BP	HB-BP-86	698	4 K 26 HB-BF-86
Size	6 in.	-	6 in.	3 in.	4 MG DW 6	-	4 in.
Relief Cap.	345,000 lb/hr	288,000 lb/hr	420,000 lb/hr	293,300 lb/hr	Unit 2 408,000/lb/hr Unit 3 420,000 lbhr	345,000 lb/hr	283,430 lb/hr
Set Press. (Reseat Press.)	2485 psig	2485 psig 2-5% blowdown	2485 psig 2485 psig	2485, 2535, 2585 ~2585, 2335, 2585	2585 psig* 2360 psig*	2485 psig	2485 psig 2410.45 psig
Known malf. (significant)	None	None	None	None	None	None	None
Power-Operated Relief Valves							
MFG	Copes-Vulcan	Copes-Vulcan	Copes-Vulcan	Copes-Vulcan	Copes-Vulcan	Masoneilan	Copes-Vulcan
Number	2	2	2	2	2	3	2
Type							
Model no.	D-100-160	D-100-160	D-100-160	D-100-160	D-100-160	38-20771	5-121642
Size	2½ in.	2½ in.	2½ in.	3 in	2 in.	-	3 in.
Relief cap.	210,000 lb/hr	210,000 lb/hr	210,000 lb/hr	210,000 lb/hr	169,000 lb/hr	210,000 lb/hr	210,000 lb/hr

*NOTE: New valves (6/79) have CV 50 Flow = 217,000 lb/hr.

TABLE VI-1 (Continued)

CODE SAFETY-RELIEF VALVES & POWER-OPERATED
RELIEF VALVES ON PRESSURIZER

	Plant Name						
	Farley 1	H. B. Robinson	Zion 1&2	Haddam Neck	Indian Point 2&3	Beaver Valley 1	Turkey Point 3&4
Set Press.*	2335 psig	2335 psig	2335 psig	2270 psig	2335 psig	2335 psig	2335 psig
Reseat Press.	2315 psig	-	-	~2250 psig	2320 psig	2327 psig	2274 psig
Malf. Date	None	None	None	None	None	None	None

NOTES
1. At least one PORV on most plants is also set to open on rate of pressure increase (see Table IV-1 in Appendix IV).

TABLE VI-1 (Continued)

	Plant Name						
	D. C. Cook 1&2	Prairie Isl. 1&2	Trojan	Salem 1*	Ginna	San Onofre 1	Surry 1&2
Code Safety-Relief Valves							
MFG	Crosby	Crosby	Crosby	Crosby	Crosby	Crosby	Crosby
Number	3	2	3	3	2	2	3
Type	E	E	E		E	HB	
Model No.	HB-BP-86	HB-BP-86	HB-BP-06E	HB-BP-86	HB-BP-86		HB-86-BP-E
Size	6M6	6 in	6M6	6M6	4K26		6K2-6
Relief Capacity	420,000 lb/hr	345,000 lb/hr	420,000 lb/hr	288,0001 lb/hr	2485 psig		293,330 lb/hr
Set Pressure	2485 psig	2485 ± 1% psig	2485 psig	2485 psig	2485 psig	2500 and 2525 psia	2485 psig ± 1%
Reseat Pressure (approx.)	2360 psig	2360 ± psig	2361 psig	2360 psig	~2386 psig	-	2360 psig ± 1%
Known Malfunctions (significant)	None	None	None	None	None	None	None
Power Operated Relief Valves							
MFG	Masoneilan	Copes-Vulcan	Copes-Vulcan	Copes-Vulcan	Copes-Vulcan	ACF Industries	Copes-Vulcan
Number	3	2	2	2	2	2	2
Type	38-20721	D-100-160	D-100-160	D-100-160	D-100-160	70-18-9 DRTX	IA 58 RGP
Size	-	2½ in	3 in	3 in	2½ in	-	2½ in
Relief Capacity	210,000 lb/hr	179,000 lb/hr	210,000 lb/hr	210,000 lb/hr	179,000 lb/hr	108,000 lb/hr	210,000 lb/hr

TABLE VI-1 (Continued)

	Plant Name						
	D. C. Cook 1&2	Prairie Isl. 1&2	Trojan	Salem 1*	Ginna	San Onofre 1	Surry 1&2
Set Pressure*	2335 psig	2335 psig	2350 psig	1 PORV at 2335	2335 psig	1 PORV at	2335 psig
Reset Pressure	3215 psig	2335 psig	2315 psig	1 PORV at 2315	2185 psig	~2190 psig	2255 psig
Malfunction Date (significant) cause	Unit 2 9/22/78 did not open at normal setpoints	Unit 2 3/22/74 below normal setpoints Unit 2-7/78 transmitter failure	None	None		None	None

*NOTE: At least one PORV cn most plants is also set to open on rate of pressure increase (see Table IV-1 in Appendix IV).

TABLE VI-1 (Continued)

Valves	North Anna 1	Point Beach 1&2	Kewaunee	Yankee Rowe
Code Safety Relief Valves				
MFG	Dresser	Crosby	Crosby-Ashton	Manning/Maxwell & Moore
Number	3	2	2	2
Type				
Model No.	6RV58LS8	HBBP86	HB-BP86	1719WA
Size	2½	4-KZ-6	6 in.	2 in
Relief Capacity	380,000 lb/hr	288,000 lb/hr	345,000 lb/hr	42,000 and 84,700 lb/hr
Set Pressure	2485 psig ± 1%	2485 psig	2485 psig	2485 psig and 2560 psig
(approx.)	code allowable	2360 psig	~2385 psig	2385 psig and 2475 psig
Known Malfunctions (significant)	None	None	None	None
Power Operated Relief Valves				
MFG	Masoneilan	Copes-Vulcan	Copes-Vulcan	Dresser
Number	2	2	2	1
Type	Globe			
Model No.	38-20721	D-100-160	D-100-160	31533 VX
Size	2.07 in	2½	2½	2½
Relief Capacity	210,000 lb/hr	179,000 lb/hr	175,000 lb/hr	70,506 lb/hr
Set Pressure*	2335 psig	2335 psig	2335 psig	2400 psig
Reseat Pressure	2315 psig	2315 psig	2330 psig	2350 psig
Malf. Date (significant)	None	None	None	None
Cause				

*At least one PORV on most plants is also set to open on rate of pressure increase (see Table IV-1 in Appendix IV).

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There are two separate controls for the PORVs. One valve is controlled by a bistable in a pressure measurement channel. When the primary system pressure exceeds a given pressure setpoint, the PORV lifts and will not reseat until the pressure has been reduced to below the bistable hysteresis point. This second PORV (three valves are installed in plants with 100% load rejection capability) is controlled by a pressure control system. Often this system has reset, and derivative gains have resulted in an output that anticipated the actual pressure. This "corrected" pressure can result in a substantial reduction in the actual PORV opening point for transients which have a high rate of pressure increase, or which require sustained operation of the pressurizer spray valves.

All PORVs have motor-operated block valves located upstream of the PORV, so that a PORV can be isolated if seat leakage becomes significant. In fact, some plants have operated for extended periods with one or all valves isolated for this reason. The block valve can also be used to isolate the PORV, should it open and fail to reclose. Indications are provided in the control room for the operator to detect the failure of a PORV to reclose. These indications include temperature detection on the discharge pipe, position indication of the PORV represented by lights, and pressure relief tank temperature, pressure and level. The response of the temperature detector does not always indicate valve position failure promptly, because of the time lag in cooling off the discharge piping after PORV closure. Conversations with W indicate that the position indication of the PORV in the control room is direct, that is, it is derived from input switches on the relief valve, thereby providing the operator with the actual physical position of the valve. Although our survey of operating W-designed reactors appears to support this statement, no attempt was made during our review to confirm this part of the PORV design. This matter has been addressed by the requirement specified in NUREG-0578 for direct indication of relief and safety valves.

1.2 Performance

Data from operating W-designed plants in the United States shows that the PORVs have opened approximately 60 times during normal operation for various reasons. For each of these openings, the valve reseated correctly. In most cases involving PORV openings, the data which we were provided did not include the preoperational testing phase of operation. The failure of a PORV to reseat fully was recently reported at the McGuire, Unit 1, plant (Duke Power Company is the owner of this facility, which does not yet have an operating license), which was performing hot functional testing. The malfunction was the result of the valve plug binding in the valve bonnet recess area. The PORV installed at the McGuire plant is of a different design than the PORVs installed at all operating W-designed plants. The exclusion of this type of data from the data base affects the calculated frequency of a PORV failure to reclose. Without additional data from all the W-designed plants on the substantial number of challenges to PORVs occurring during such testing and any associated failures to reclose, the failure rate during preoperational testing cannot be tabulated and factored into the overall PORV failure rate.

We are aware of an event at the Beznau reactor in Switzerland, which was designed by W, that occurred five years ago, in which a PORV was challenged during a turbine trip transient, and failed to reclose when the primary system pressure decreased. The failure to close was detected in a few minutes by the operators, who immediately isolated the valve by closing the block valve located in series with the PORV. This action terminated the incident. The failure to reclose was due to the rupture of the cast-iron frame between the valve operator and the valve body. This rupture was caused by a water slug hitting the valve. The source of the water slug was the loop seal located between the pressurizer and the PORV. Investigation of this event identified the cause of the valve failure as design error, which we understand has been subsequently remedied. We believe that this event is significant with respect to our review of PORV experience at W-designed operating reactors in the United States and must be considered as a relevant statistic.

At an April 23, 1979 meeting with the NRC staff, W representatives stated that the following events could cause opening of the PORV:

(1) Rod withdrawal from low power.

(2) Loss of offsite power.

(3) Turbine governor or control valve closure.

(4) Main steam isolation valve closure.

This conclusion is based upon best estimate calculations performed by Westinghouse (see Tables VI-2 and VI-3 for summaries of assumptions and results, respectively). Westinghouse stated that, based upon plant operating experience, they actually did not expect PORV openings on a loss of offsite power. The preliminary data we have received appears to support this conclusion.

1.3 Transients That Lift PORVs

Westinghouse pressurizer PORVs are designed to prevent challenges to the pressurizer safety valves and to keep the units on line. In part, this design philosophy is exemplified by setting the PORV open signal at a lower pressure than the high pressurizer pressure reactor scram setpoint. This protects the pressurizer safety valves and increases the load rejection capability of W-designed reactor units. However, it also has the potential for generating additional PORV challenges.

Final Safety Analysis Reports (FSARs) for W-designed operating plants indicate that pressurizer PORVs are only challenged during the course of a few transients. Conservative analyses presented in the FSARs generally indicate that loss-of-offsite power, uncontrolled rod withdrawal at low power, turbine governor or control valve closure, main steam isolation valve closure, and loss of load will cause the pressurizer PORVs to open. These conservative analyses also indicate that certain small break LOCAs or loss of secondary heat removal events coupled with a single failure in the auxiliary feedwater system may challenge the PORVs.

TABLE VI-2

TRANSIENT ANALYSIS ASSUMPTIONS:

(BEST ESTIMATE VS FSAR)

Intitial Conditions	Best Estimate	FSAR
Power	Rated	E_SF + Calorimetric
Pressure	P_{nom}	$P_{nom} \pm 30$
Temperature	T_{avgnom}	$Tavg_{nom} \pm 4$
Steam generator level	L_{nom}	$L_{nom} \pm 5\%$ level error $\pm 10\%$ mass uncertainty error
Rod Drop Time	BE flow	TD flow and uncertainites
RCS flow	BE	TDF
Decay heat	ANS	Appendix K
Reactivity coefficents	BE	Bounding
Trip setpoints	PLS	PLS ± instrument errors
Auxiliary feedwater Flow	No single Active failure	Worst single active Failure assumed
Safety Injection flow	Maximum safeguards flow	Minimum safeguards flow

TABLE VI-3

TRANSIENT ANALYSIS RESULTS USING ASSUMPTIONS IDENTIFIED IN TABLE VI-2

Accident	PORV opening	
	Best Estimate	FSAR
Loss of offsite power	No	Yes
Loss of normal feedwater	No	Yes
Loss of load		
Turbine trip	No	No
MSIV closure	Yes	Yes
Turbine valve closure	Yes	Yes
Feedline rupture	Yes	Yes
Steamline rupture	Yes	Yes

However, most transients that actually occur in nuclear power plants are not as severe as those postulated in FSARs (e.g., less limiting initial conditions or system failures, or non-biased heat transfer coefficients). Westinghouse indicates that, among the transients analyzed in FSARs, only uncontrolled rod withdrawal at low power, turbine governor or control valve closure, and mainsteam isolation valve closure should actually life a PORV. The discussions contained in the next two sections describe the loss of load and rod withdrawal transients as analyzed in the FSARs.

1.4 Loss of Load Transient (As Analyzed in FSARs)

Most overpressure transients, such as loss of load, generate high pressures in the primary system by terminating or restricting the heat removal capacity of the secondary system. Loss of load on a typical W-designed unit restricts secondary heat removal by tripping the turbine, closing the turbine flow control and stop valves and by opening the turbine bypass valves, the atmospheric steam dump valves, and possibly the steam generator safety valves. The reactor does not trip on turbine trip (since this is an anticipatory trip - see Table V-2 in Appendix V) for most W-designed plants and the core power remains essentially unchanged. The turbine bypass valves and steam generator atmospheric dump valves are not capable of completely dissipating the energy generated in the core at full power. The secondary side pressure hangs up at the relief pressure of the atmospheric dump valves or the steam generator safety valves, thereby degrading the heat transfer coefficent from the primary system to the secondary system and increasing the primary system temperature. The primary system heats up, expands and increases in pressure. At about 2335 psig, the pressurizer PORVs open to help reduce primary system pressure. A few W-designed operating plants are designed to accept a high load rejection, some as high as 100 percent of full load. For these plants, the pressure should not reach the high pressurizer pressure reactor trip setpoint of about 2385 psig. At other W-designed units, the pressure will exceed the high pressurizer pressure reactor trip setpoint and scram the reactor. After reactor scram, core power is reduced to decay heat levels and primary system pressure is reduced while the PORV's close at their reseat pressure.

1.5 Rod Withdrawal Transient (As Analyzed in FSARs)

In the case of an uncontrolled rod withdrawal, additional power is produced in the core region where the rod is withdrawn. If the rod has low worth or is withdrawn slowly, the reactor will trip on overpower differential temperature. The turbine sees no demand for additional power, so the secondary system continues to remove a constant amount of energy for the primary system. Eventually the primary system will heat up, expand, and the pressure will increase as the core continues to generate energy in excess of that removed by the secondary system. The primary system pressure increases to about 2335 psig where the pressurizer PORVs open to relieve system pressure. The reactor scrams due to a high Differential temperature across the core. Pressure and system energy decrease until the PORVs close at their reseat pressure. If the withdrawing rod has a high rod worth or if it is withdrawn rapidly, the reactor will scram instead on high flux.

Table VI-4 is a list of all operating W-designed plants in the United States with the PORV openings and reason associated with each. These data were obtained from a recent survey of W-designed plants, and were derived in some cases from plant recores and in others from the utilities' and operators' recollection. A summary of the Table VI-4 data is provided in Tables VI-5 and VI-6.

Of the 26 events listed in Table VI-3, six openings at Point Beach and Haddam Neck occurred more than five years ago and have not occurred since (supposedly due to plant modifications or changes in procedures). This leaves at least 20 events which may be of the type which could occur again.

A large fraction of these events are due to intentional testing during the preoperational testing period, startup testing or prior to refueling (for low temperature overpressure protection). The data indicate that many openings are due to operator error while conducting tests or maintenance. Several of these were at significant power levels.

1.6 Alternatives Considered to Reduce Challenges to the PORV

At our request, W investigated the possibility of changes in the PORV setpoint and high pressure reactor trip setpoint to reduce or prevent PORV openings for transients. W noted that the PORV is designed to prevent safety valve operation and would therefore have to be set below the safety valve setpoint by a certain margin. W states that for a main steam isolation valve closure event, the peak pressure is calculated to be 140 psi above the scram setpoint. If the scram and PORV setpoints are to be modified to prevent PORV opening for this event, the adjustment of reactor trip setpoint, they contend, would cause additional spurious reactor trips (See Figure VI-1). The W position is that , even if this were successful, less than half of all PORV openings reported would be precluded. It is therefore not apparent that this reduction is beneficial when compared to an increased number of spurious high pressure reactor trips. Further discussion of this matter and the actions that we have recommended be implemented regarding the reduction of the frequency of occurrence of PORV failure (to close) may be found in Section 3 of Appendix VIII, which addresses the frequency of small break LOCAS.

1.7 Liquid and Two-Phase Relief

Pressurized water reactor (PWR) relief and safety valves are designed to discharge saturated steam. Analyses for anticipated transients, other than a stuck-open PORV, do not consider the filling of the pressurizer, which would result in two-phase or liquid relief. However, for a stuck-open PORV, the pressurizer will fill and the PORV will experience two-phase flow conditions.

The event at the Beznau reactor in Switzerland in which PORV damage possibly resulted from water slug relief suggests that water relief through either this type of PORV or safety-relief valves may not be well understood. Westinghouse, however, states that, although PORVs are expected only to relieve steam, they are nevertheless capable of passing two-phase and liquid water and no destructive failure of the valve body will occur for any deterioration of valve

TABLE VI-4
PORV OPENING DATA FOR ALL OPERATING W-DESIGNED PLANTS

Facility	No. of PORV Openings	Reason/No of Incidents
Beaver Valley 1	13	Note 2/1 Note 3/10 Note 5/1 Note 6/1
D.C. Cook 1&2	5	Note 1/1 Note 2/1 Note 3/2 Note 4/1
Farley 1	3	Note 3/3
Indian Point 2&3	0	
Kewaunee	2	Note 1/2
North Anna 1	0	
Point Beach 1&2	5	Note 1/2
Prairie Island 1&2	3	Note 1/1 Note 2/1 Note 3/1
H. B Robinson 2	1	Note 4/1
Salem 1	3	Note 4/3
Surry 1&2	4-6	Note 2 4/6
Trojan	1	Note 3/1
Turkey Point 3&4	10*	Note 1/1 Note 2/7 Note 3/2
Zion 1&2	3	Note 5/1 Note 1/1 Note 8/1 Note 3/1
Haddam Neck	3	Note 3/3
Ginna	0	
San Onofre 1	1	Note 1/1
Yankee Rowe	0	Note 7
Total	57-59	

Note 1: Instrumentation or technician/error
Note 2: Intentional test
Note 3: Transient response
Note 4: Cold shutdown, water solid
Note 5: Manual opening to control pressure on transient
Note 6: Cause unknown
Note 7: No automatic, several manual openings
Note 8: Loss of pressurizer spray flow. PORVs opened. Isolation valves shut, therefore, valves did not relieve.

*Licensee states that numbers are approximate

TABLE VI-5

SUMMARY OF PORV OPENING DATA

Cause for PORV Opening	Plants
Instrument or technical error	9
Intentional opening for test	16
Intentional opening for pressure control	2*
Cold shutdown, water solid	5
Cause unkown	1
Transient-automatic response	26

Note: All valves closed normally.

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*Yankee Rowe reports several manual openings but no reason given for opening.

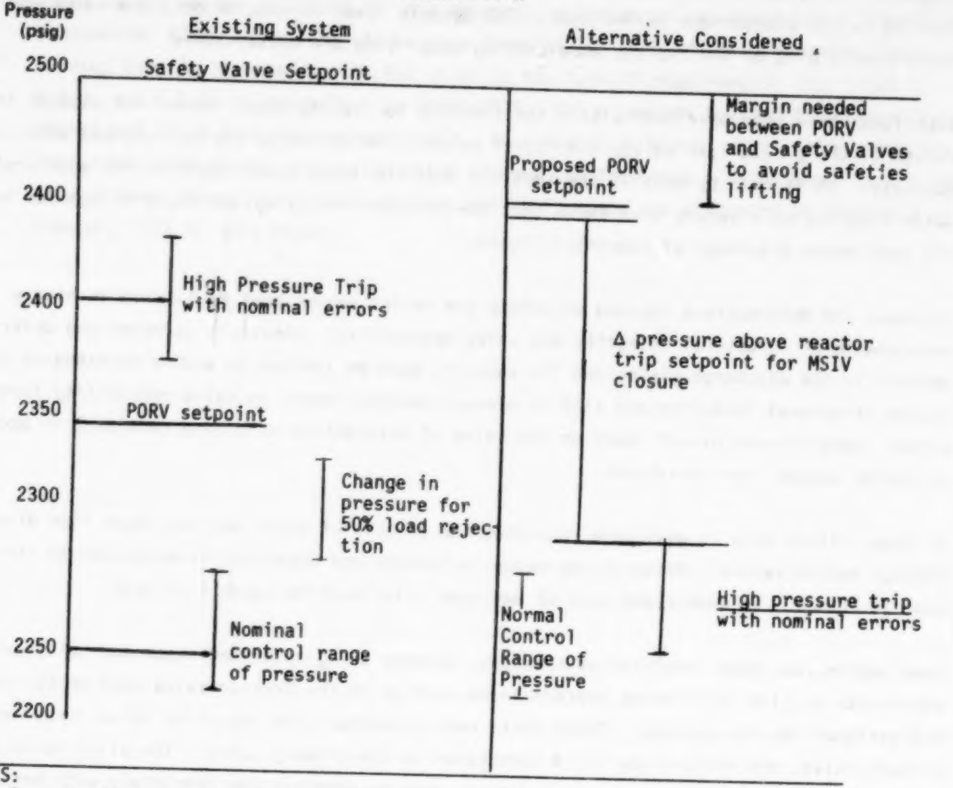
TABLE VI-6

SUMMARY OF AUTOMATIC OPENINGS DURING TRANSIENTS

Facility	Cause
Prairie Island 1&2	Maintenance being performed on electro-hydraulic control oil system caused a turbine trip. Control rods were in manual mode. Pressure increased to 2285 psig.
Point Beach 1&2	Main steam isolation valve closure (3 times).
Turkey Point 3&4	Full load rejection (2 times).
D.C. Cook 1&2	Failure of crids. I and II vital power supply inverters; crids with safety injection signal. Both resulted in reactor trip and loss of pressurizer spray.
Beaver Valley 1	Turbine trip, rapid load reduction, main steam isolation valve closure, rapid load decrease.
Trojan	High steam generator water level caused feedwater isolation and turbine trip/reactor trip.
Haddam Neck	Loss of ac power (3 times 1968-1970).
Farley 1	Loss of main feedwater at 87% power with pressure control in manual.
	Reactor trip during startup with pressure control in manual.
	Loss of circulating water and loss of 4160-volt bus during lightning storm caused safety injection, and overfilling of pressurizer. PORVs opened and shut about 4 times.
Zion 2	Reactor trip and safety injection from 20% power. Inspection of containment revealed pressurizer relief tank rupture disks blown due to apparent opening of PORVs and safeties.

COMPARISON OF SETPOINTS

FIGURE VI-1.


Pressure (psig)
Existing System
Alternative Considered
2500
Safety Valve Setpoint
Margin needed between PORV and Safety Valves to avoid safeties lifting
Proposed PORV setpoint
2400
High Pressure Trip with nominal errors
2400
Δ pressure above reactor trip setpoint for MSIV closure
2350
PORV setpoint
2300
Change in pressure for 50% load rejection
High pressure trip with nominal errors
Normal Control Range of Pressure
Nominal control range of pressure
2250
2200

NOTES:

1. Nominal error in pressure channel ± 25 psi
2. Nominal error in high pressure trip channel 3% of span or ± 25 psi
3. Pressure increase for
   a. 50% load rejection + 60 psi
   b. 100% load rejection + 110 psi
   c. MSIV closure + 150 psi above high pressure trip setpoint

Conclusion

Proposed system will result in a reactor trip at full power seating surfaces. Typically during a plant transient, if the primary system reaches the PORV setpoint, the slug of water is propelled by steam through the valve and into the piping leading to the pressurizer relief tank. The dynamic loads imposed on both the valve and the downstream piping by the rapidly accelerating water slug are quite severe.

Techniques have been developed within the industry for taking these loads into account in the design of the pressurizer safety and relief valves, the discharge piping, and the associated supports. It is felt by many in the industry that the loads resulting from the accelerating water slug are more severe than those that the pressure relief system would be exposed to by the continuous discharge of subcooled liquid.

Stresses and deformations imposed on safety and relief valves must be limited to assure maintenance of structural integrity and valve operability. Similarly stresses and deformations in the discharge piping and its supports must be limited to assure maintenance of piping structural integrity and also to prevent adverse impact on valve operability through either imposition of direct loads on the valve of deformation of piping resulting in excess discharge system flow resistance.

To date, little data is available regarding the effects of water and two-phase flow discharge through relief valves. Research has begun in Germany and Japan and is scheduled to start soon in France. We understand that CE may have valve testing capability soon.

Some testing has been completed at Erlangen, Germany using saturated steam and hot pressurized water with a pilot valve which appears to be similar to the Dresser valve used on CE- and B&W-designed reactor systems. These tests were conducted with the pilot valve connected to a primary valve, but without any fluid connection to the primary valve. The pilot valve performe as expected with saturated steam, but did not open as smoothly and completely with hot, pressurized water.

The French intend to test a six-inch, self-activated spring-loaded safety valve, while the Japanese have scheduled tests for what appears to be boiling water reactor safety-relief valves.

An additional concern is related to the operability and relief capacity of the PORVs and the spring-actuated safety valves, in that the valves currently installed on operating reactors were designed to perform their pressure relieving function using saturated steam as the working fluid. This is especially a concern as it relates to the mitigation of potential anticipated transients without scram (ATWS) and other overpressure events where two-phase and subcooled water discharges are expected to occur. This concern has been addressed by the requirements specified in NUREG-0578 regarding testing of safety and relief valves.

2. CONCLUSIONS

Based on the review described in this appendix, a number of recommendations regarding pressurizer PORVs and safety valves were formulated. Some of these recommendations have already been implemented by the NRC staff in the form of requirements (see Items 2.1.1, 2.1.3, and 2.1.3a in Appendix A to NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations"). In addition to these, other recommendations were identified during our review which address the frequency of challenges to the PORVs. As mentioned above, the recommendations and their bases are discussed in Section 3 of Appendix VIII of this report

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APPENDIX VII

NATURAL CIRCULATION IN WESTINGHOUSE-DESIGNED OPERATING PLANTS

1. GENERAL FEATURES

Under normal subcooled conditions in the primary system of Westinghouse (W)-designed operating plants, natural circulation is maintained by the density gradient between the core side leg and the steam generator primary leg. Generally in W designs, the bottom of the steam generator tube sheet is approximately 18 feet above the top of the active fuel. This elevation difference provides a driving force for maintaining natural circulation. A discussion of this mode of cooling for various conditions and related data are provided in the following paragraphs.

1.1 Natural Circulation Testing

Tests have been conducted on a number of W-designed operating plants to demonstrate system performance during natural circulation. The test results for Point Beach (two loops, 1518 MWt), Zion Unit 1 (four loops, 3250 MWt), and Haddam Neck (four loops, 1825 MWt) are discussed below. Apparently, W does not recommend a natural circulation test for each reactor, but believes that first-of-a-kind testing is sufficient to verify the natural circulation capabilities(1) for each class of reactors. Each plant discussed below has conducted a natural circulation test to verify proper system performance. The results from the tests generally show that the initial core temperature differential increased for about 15 minutes of the test, then then decreased to a steady state value for the remainder of the test. Figure VII-1 was presented to the staff during an April 26, 1979, meeting. This curve shows the results of several W natural circulation tests.

1.1.1 Point Beach

The natural circulation test was accomplished by securing both reactor coolant pumps and allowing the density difference between the cold coolant in the cold leg and reactor vessel downcomer and the hot coolant at the top of the core and in the hot leg to become the driving force in removing heat from the core. There were three ways in which natural circulation was verified: circulation indicated on the flow instrumentation, a temperature difference indicated by the hot leg and cold leg resistance temperature devices (RTDs), or stabilization by the incore thermocouples at a particular temperature. The pumps were secured at essentially zero power. The reactor thermal output was increased by control rod withdrawal. The amount of heat removed from the steam generators was determined by measuring the "boildown" rate. Feedwater flow to both steam generators was secured when they indicated a relatively high water level. Steam was being dumped through the atmospheric dump valves as a means of pressure control.

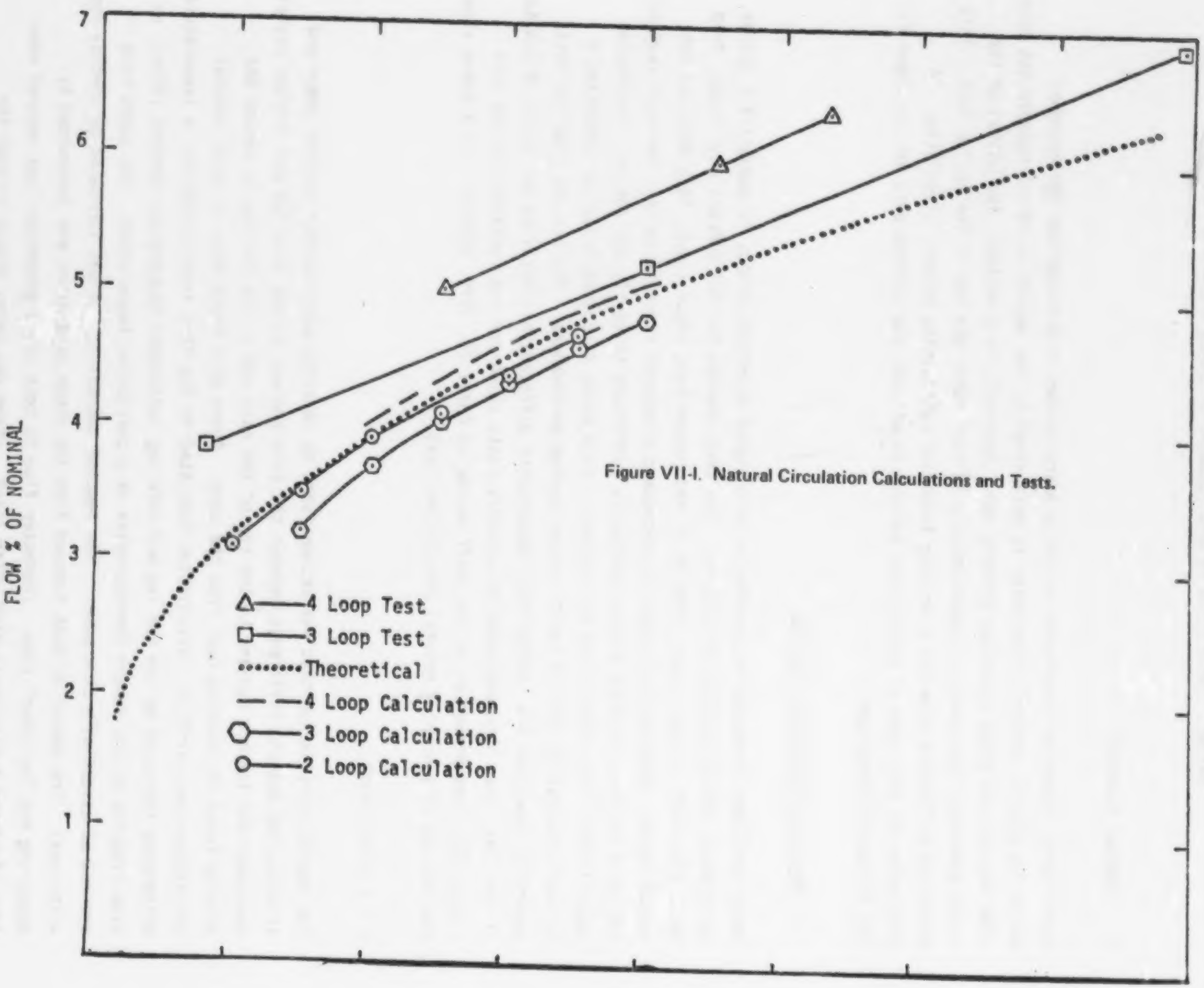

Figure VII-1. Natural Circulation Calculations and Tests.

The incore thermocouples were the best indicators of natural circulation. When the power increases were stopped, the thermocouple readings would increase to a new value and stabilize. At approximately 6% power, the average thermocouple temperature was 600°F. The table below summarizes the test results.

	Loop A		Loop B	
Core Power	ΔT	%Flow	ΔT	% Flow
34.6 MWt (~2.39%)	13°F	13.7%	17°F	5.8%
63.2 MWt(~4.2%)	19°F	15.3%	22°F	9.5%

1.1.2 Haddam Neck

The test was conducted by tripping all reactor coolant pumps from a hot standby condition (reactor shutdown). The reactor had just been shut down (approximately one hour earlier), following a month of operation at about 70% power. The steam generators were initially filled to a relatively high water level. When the test was initiated, the steam generator water level was allowed to drop, due to boiloff from energy removal. When the water level reached a predetermined minimum, feedwater flow was initiated, and the steam generator water level was rapidly restored to its initial value. The resulting cooldown of the reactor coolant system (from the addition of feedwater) gave a measure of loop transfer time by tracking the "cold slug." Two methods of measuring natural circulation were used in the test: the loop transport time method utilizing the ΔT between the time the "cold slug" was felt at the cold leg temperature (T_C) RTD and the hot leg temperature (T_H) RTD. The other method was an analytical method utilizing calculated decay heat, known piping loss coefficients and heat losses, and measured ΔTs. As can be seen below, both methods gave reasonably consistent results.

(1) Loop Transport Time Method: 3.18% flow (each loop)
(2) Analytical Method: 3.29% flow (each loop)

1.1.3 Zion Unit 1

With the reactor critical and at a steady state condition of approximately 2% reactor power, the reactor coolant pumps were tripped. When steady state conditions were reached, data were collected. The steam generator water level was maintained by continuous feeding with the auxiliary feedwater system. After data were collected, reactor power was increased in 2% steps to 6% and data collected following each step power change, after steady state conditions were achieved (approximately 10 minutes). Main condenser hotwell fluid level and secondary storage tank water level were monitored to aid in the determination of steady state conditions. After all test data had been collected, the reactor was brought subcritical and the control rods were inserted to permit a cooldown rate of 50°F/hr. During the test, none of the temperature limitations for core thermocouples, loop ΔT, or average coolant temperature were exceeded.(1)

By extrapolating the data, a reactor power of approximately 7% could be achieved by natural circulation before the core exit thermocouple temperatures reached an unacceptable value (~620°F). The measured reactor coolant system flow during the test was comparable to earlier predictions in the Final Safety Analysis Report (3.28% flow at 1% power, 5.20% flow at 4.0% power).

Figure VII-I shows the primary system trends for a typical natural circulation test.

There have been many sustained loss of offsite power events resulting in natural circulation in W-designed plants. The staff has conducted a survey of W-designed operating reactors to confirm that natural circulation cooling occurred as designed after the loss of power, or the loss of reactor coolant pumps for other reasons. The results of this survey show that, for each of the 40 reported events, natural circulation was reported to be successful (see Tables VII-1 and VII-2). In three instances, a natural circulation cooldown was conducted to the point where the residual heat removal (RHR) system could be used for cooling.

1.2 Primary System Pressure Control

Reactor coolant system pressure control is necessary to maintain an adequate subcooling margin and assure no disruption of the natural circulation flow. Once natural circulation is achieved, system pressure control would be accomplished by maintaining a system overpressure with the pressurizer through use of the pressurizer heaters. The pressurizer heaters are available for this condition, since the need for natural circulation cooling was initiated by a complete loss of flow transient. The loss of offsite power would also initiate a complete loss of flow transient. However, in this case, the pressurizer heaters are generally not immediately available, since some of them are not on the onsite emergency buses and cannot be manually loaded on the buses. During this condition, system pressure control is accomplished by:

(1) Controlling the system temperature by controlling the rate of energy removal from the primary system by the steam generator, or

(2) Controlling the liquid level in the pressurizer to account for the cooling off of the liquid, steam, and metal.

Eventually, if the pressurizer heaters are not restored, the pressurizer must be taken water-solid to control system pressure, since the pressurizer itself cools off due to ambient heat losses. Westinghouse states that the heat losses to ambient are small and recommends to their customers that the pressurizer heaters should have backup capability within three hours powered from a vital power supply. We conducted a survey of all W-designed operating reactors to evaluate the pressurizer heater capability without offsite power available. The results of the survey are given in Table VII-3. These results indicate that there are several reactors which at the time the survey was taken, did not have Class 1E backup heater capability.

TABLE VII-1 NATURAL CIRCULATION IN W-DESIGNED OPERATING PLANTS

Plant	Power Level (MWt)	Number of Loops	Event Description/ Number of Occurrences	Total
Farley 1	2652	3	All reported events mode 4,5,6	0
H.B. Robinson	2200	3	Note 2-1, Note 3-1, also 7	2
Zion 1&2	3250	4	Note 2-1	1
Connecticut Yankee	1825	4	Note 2-1, Note 4-1	2
Indian Point 2&3	2758/3025	4	Note 2-1	1
Beaver Valley 1	2652	3	Note 2-2, Note 4-4	6
Turkey Point 3&4	2200	3	Note 4-9	9
D.C. Cook 1&2	3280/3391	4	Note 5-5, Note 6-1	6
Prairie Island 1&2	1650	2	Note 2-1	1
Trojan	3411	4		1
Salem 1	3338	4	None	0
Ginna	1520	2	Note 5-1, Note 2-1	2
San Onofre 1	1347	3	Note 2-1, Note 5-1	2
Surry 1&2	2441	3	None	0
North Anna 1	2775	3	Note 7-1	1
Point Beach 1&2	1518	2	Note 4-3	3
Kewaunee	1650	2	Note 8-2	2
Yankee Rowe	600	4	Note 2-1	1
			Total:	40

NOTES

1. Data does not include periods when reactor coolant pumps were lost in hot standby on transition to RHR mode (mode 3 to mode 4) or loss of RHR during mode 4,5 or 6*.
2. Preoperational or Startup Test
3. Reactor coolant pump seal failure at power
4. Loss of ac power (loss offsite power) - power level not known
5. Loss of power or reactor coolant pumps at full power
6. Loss of power or reactor coolant pumps at low power
7. Steam dump valve stuck open initiated ESFAS, pressurizer level low, operator secured reactor coolant pumps
8. Natural circulation cooldown to RHR cut-in
9. Two plants have not yet submitted data.

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*	
Mode 1	Power Operation
Mode 2	Startup
Mode 3	Hot Standby
Mode 4	Hot Shutdown
Mode 5	Cold Shutdown
Mode 6	Refueling

TABLE VII-2
NATURAL CIRCULATION DATA SUMMARY

Data Summary	Plants
Preoperational of startup tests	11
Loss of ac power reactor coolant pumps at full power	8
Loss of ac power/reactor coolant pumps at less than full power/power level not known	17
Natural circulation cooldown to RHR cut-in	3
Operator secured reactor coolant pumps on low pressurizer level after steam dump valve failure	1
Total =	40

TABLE VII-3

BACKUP CLASS 1E POWER SUPPLY FOR PRESSURIZER HEATERS

Plant Name	Backup Heater	Proportional Heater
Beaver Valley	YES	
D.C. Cook 1 & 2	NO	
Farley	YES	NO
Ginna	YES	
Haddam Neck	NO	
H. B. Robinson	YES	
Indian Point 2	YES	
Indian Point 3	*	
Kewaunee	YES	
North Anna 1	YES	
Units 1 & 2		
Point Beach 1 & 2	YES	
Prairie Island 1 & 2	NO	YES
Salem 1	*	
San Onofre 1	YES	
Surry	YES	
Trojan	YES	
Turkey Point 3 & 4	YES	
Yankee Rowe	NO	
Zion	YES	

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*Information not available at time of report.

The importance of subcooling during natural circulation and the role of the pressurizer heaters have been discussed above. However, the necessity of pressure control using the pressurizer, and thereby the pressurizer heaters, during long-term cooling following a small break loss-of-coolant accident (LOCA), with and without offsite power available, has not been addressed by the licensees. It appears that isolation of a small break, if possible, and/or the return to a hot standby cooling mode with a bubble in the pressurizer may be desirable, and the pressurizer heaters would aid in achieving the needed pressure control. As observed at Three Mile Island Unit 2, the pressurizer heaters were lost due to moisture and electrical grounds and subsequently regained. This points out that the availability of the pressurizer heaters after a small break LOCA will depend on the control and power system environmental qualification for such an event. The level of qualification on currently operating reactors has not been evaluated. NUREG-0578 has specified requirements for the emergency power supply for the pressurizer heaters which address this concern.

1.3 Natural Circulation Cooling

The reactor trips on a loss of flow signal when the reactor coolant pumps have tripped off line.

The steam generator level is initially maintained constant by the auxiliary feedwater system. When the reactor coolant pump and reactor trips occur, the steam generator water level drops, due to the collapse of voids from the loss of load. The level is slowly recovered using manual or automatic control of the auxiliary feedwater system. Once the steam generator water level is at the preset value, feedwater flow is terminated. The core ΔT initially decreases due to the reactor trip, and the hot leg temperature (T_H) initially drops, then rises up to slightly below the initial T_H. This is significant, since the primary system pressure may be dropping and T_H may approach the saturation temperature. Natural circulation is achieved when the ΔT across the core indicates that decay heat is being removed, and the steam generator water level has been restored to provide the elevation difference necessary for the driving force. This elevation difference, in combination with the reactor coolant system pressure control (discussed in the preceding paragraphs), assures that natural circulation cooling is maintained.

1.4 Natural Circulation Cooling During Loss of Offsite Power (LOOP)

The W-designed plants would be expected to enter a natural circulation condition following a loss of offsite power without any operator action. However, operator action is eventually required to control steam generator water level and reactor coolant system pressure.

During a loss of offsite power, the reactor would automatically trip, and the turbine would trip on either loss of load or loss of condenser vacuum. The main feedwater pumps would also be tripped, since their turbines would lose their exhaust (main condenser). The auxiliary feedwater system is automatically initiated on most of the W-designed plants, so makeup is available without manual operator action being taken. For plants which require manual action to initiate the auxiliary feedwater system, we have recommended (see Appendix III of this report) addition of circuitry to automatically initiate the auxiliary feedwater system.

The steam generator inventory is conserved, since the reactor is shut down early, thus avoiding significant reactor coolant system mismatches (heat input-heat output). As the reactor coolant pumps coast down and feedwater flow drops, the reactor coolant system heats up to develop the core ΔT necessary to support natural circulation. The pressurizer fluid level initially drops (rapidly) due to the reactor coolant volume contraction, then, as the reactor coolant system heats up, the level rises. The W FSAR-type analyses predict this later insurge to result in lifting of the PORV(s), but experience has shown that this may does not occur (see Sections XI.1.2 and Table VI-4 in Appendix VI for details). It is thought that the modeling of the pressurizer steam-to-wall heat transfer plays a significant role in the prediction of pressurizer pressure versus liquid level during an insurge.

The steam generator secondary side water level remains above the minimum level necessary to support heat transfer (and natural circulation) for at least 25 minutes without any auxiliary feedwater. The steam generator pressure is controlled by either the atmospheric dump valves (which may, in some plants, be automatically actuated as part of the steam dump control system) or by the Code safety valves. In either case, energy is removed from the secondary system by steaming in the steam generators. The rate of steaming is carefully controlled (if manually performed) to control the reactor coolant system temperature and pressure. As discussed in Section 2 of Appendix IV, the pressurizer pressure control system (pressurizer heaters, if available, and auxiliary spray) would be used to control the primary system pressure until the conditions have been reached where the residual heat removal system can be put into operation to achieve cold shutdown conditions. If the pressurizer heaters should not be available, it may be necessary to operate the high pressure injection (HPI) system to maintain natural circulation conditions. Implementation of the NUREG-0578(2) requirements regarding emergency power supplies for the pressurizer heaters and associated controls will provide the capability to maintain natural circulation at hot standby conditions following a loss of offsite power through the use of the pressurizer heaters without the necessity for HPI system operation.

1.5 Natural Circulation Cooling During Station ac Blackout

The W-designed plants would be expected to enter a natural circulation condition following a station blackout (loss of all offsite alternating current (ac) power and the loss of all onsite ac emergency power systems). Operator action would be required to control steam generator water level and, at some later point, the reactor coolant system pressure. The need for other operator actions is further discussed below.

Analyses of station ac blackout have neither been required by the staff nor provided by the licensees. However, as discussed previously in Section 4.2 of Appendix III, on the more recent applications for reactor licenses, the staff has been requiring that the auxiliary feedwater system be designed to withstand the complete loss of all ac power sources, including the emergency ac power source. We have not performed a complete detailed evaluation of the blackout scenario. This evaluation is expected to be performed under the Generic Task Action Plan A-44. The blackout scenario basically should follow the scenario described for the loss of offsite power, up to and including the time required for auxiliary feedwater system actuation, so that the steam generators' heat transfer capability would not be lost. This assumes that the primary coolant inventory remains the same as that for loss of offsite power, so that natural circulation can be achieved (this may not be the case but the detailed evaluation should determine that). At the time when the auxiliary feedwater (AFW) system flow is actuated/or required, the steam turbine-driven train of the AFW system is the only potentially operable train. The staff has conducted a survey of all operating W-designed plants to determine whether the AFW system is capable of performing its function under blackout condition. The results of this survey, along with the AFW system reliability analysis during a station ac blackout condition, are given in Appendix III. Most of the AFW system designs were found to have ac power dependencies, but the dependencies were ones that could be successfully overcome by local manual actions. In such designs, the reactor operator would be required to take additional manual actions to ensure the operability of the AFW system until ac power could be restored. Short-term recommendation GS-5 and long-term recommendation GL-3 in Section 5 of Appendix III of this report address our concerns associated with manual actions regarding the AFW system during a blackout condition.

Normally, the steam generator pressure can be controlled by the atmospheric dump values or, if necessary, by the steam generator safety valves. The atmospheric dump values are ac power-dependent for operation. Therefore, manual operator action at the handwheel would be required to open these valves so that the steam generator safety valves would not be required to cycle to relieve steam generator pressure.

It is not clear if the pressurizer heaters are necessary, although they would provide an easier means of pressure control. The NUREG-0578 requirements for emergency power supplies for the pressurizer heaters have addressed this concern. The instrumentation and the indications available to the operator following station ac blackout have not been addressed. As stated above, detailed evaluation of this scenario will be performed under Generic Task Action Plan A-44.

1.6 Natural Circulation Following Small LOCA

Section 4 of Appendix VIII of this report addresses the cooldown mechanism following a small break LOCA and a simultaneous loss of offsite power event.

1.7 Natural Circulation Operator Guidelines

The plant can enter the natural circulation mode of cooling in two ways. First, as a result of an upset, such as loss of power or loss of forced reactor coolant flow, or secondly, as a result of intentional operator action, which might occur in the transition from forced flow cooling to RHR system cooling. In both cases, we believe that guidelines and procedures must be provided to the operator so that sufficient cooling can be verified. Also, necessary emergency actions should be provided by procedure in case this verification shows improper plant conditions-such as loss of an adequate margin of subcooling.

Prior to the TMI-2 accident, all W guidelines (recommendations to utilities for emergency operating procedures) assumed that natural circulation was occurring. No specific guidance was provided to the operator with respect to confirmation that natural circulation was underway.

Our position concerning emergency operational procedures dealing with natural circulation was stated in our November 5, 1979 letter to the W Operating Plants Owners Group (D. F. Ross, Jr., to Cordell Reed), which transmitted our acceptance of the generic guidelines for procedures regarding small break LOCAs. We require that the emergency operating procedures include instructions for monitoring and reinitiating (if lost) natural circulation for small break LOCAs where heat removal by the steam generators is required. The generic guidelines for such procedures should direct the operator to initiate a controlled plant cooldown, if stable system conditions can be maintained. We require that each licensee provide procedures for cooling down the plant under natural circulation conditions. These procedures should address boration control and monitoring, cooldown of the pressurizer, and adequate criteria for monitoring coolant system temperatures to assure that voids do not form in the primary system which could inhibit adequate heat removal.

The W Operating Plants Owners Group is developing guidelines for emergency operational procedures regarding natural circulation as part of its effort in response to the NUREG-0578 requirements regarding inadequate core cooling. The staff's evaluation of this matter will be presented in a separate report.

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APPENDIX VIII

ANALYSIS

I. INTRODUCTION

Slow system depressurization accidents resulting from small breaks in the primary system have not, until recently, been subjected to detailed analytical study comparable to that devoted to large breaks. Typically, small breaks have been analyzed down to the smallest break size that would produce system depressurization without uncovering the core in accordance with the single failure criterion and other requirements imposed by Appendix K of 10 CFR 50. These analyses assumed the availability of heat removal through the steam generators following reactor scram, power loss to the reactor coolant pumps upon scram, and normal plant protective and emergency core cooling systems activation initiated by the system depressurization. While the analyses, in general, were sufficient to show compliance with the requirements of Section 50.46 of 10 CFR Part 50 (10 CFR §50.46), they failed to provide the necessary information needed for operator action following a small break, as was pointed out by the NRC's Lessons Learned Task Force (1).

Recent events in operating plants, including the Three Mile Island Unit 2 (TMI-2) accident, have shown that (1) an unisolated relief or safety valve failure is smaller in size than the small breaks usually analyzed in plant safety evaluations, (2) plant response to unisolated valve failures and the required operator action for such valve failures is different than for larger breaks, (3) auxiliary feedwater systems have not always performed as expected, and (4) the actual operator actions did not follow the assumptions of the safety evaluations. For example, in two out of four reported events (Davis-Besse, Oconee-3, TMI-2, and the Beznau reactor in Switzerland, which was built to a Westinghouse design), where a pressurizer power-operated pressure relief valve (PORV) was actuated by high pressure and failed in the stuck-open position, it is known that the operators prematurely terminated high pressure injection (HPI) flow on the basis of high coolant level in the pressurizer during the early part of the transient.

Each of these small break accidents was initiated by feedwater flow interruption leading to primary system pressure rise to the relief valve setpoint. In the Davis-Besse and Oconee incidents, emergency feedwater was activated as designed. Emergency feedwater response in the TMI-2 plant was delayed for eight minutes following loss of main feedwater, while emergency feedwater response in the Beznau plant was not described in available reports.

In three of the four cases of stuck-open PORVs, reactor coolant pump operation was maintained for the duration of the transient. Trip of the reactor coolant pumps occurred only in the TMI-2 case after one hour into the transient. In two of the four cases, all steam generators remained operational for the duration of the transients.

One steam generator boiled dry in the Davis-Besse event due to failure of an auxiliary feedwater system pump to come up to speed when automatically activated. One steam generator was isolated after several hours in the TMI-2 accident due to suspected tube ruptures.

Reassessment of the failure modes assumed in small break accidents as a result of these events, particularly in light of the TMI-2 accident, have led the staff and industry to a considerably broader interpretation of potential accident scenarios than held previously. This reassessment by the Bulletins and Orders Task Force has included a realistic evaluation of steam generator heat removal capability. It also considered cases where the conditions for emergency core cooling system (ECCS) pump activation were not reached, and cases where the reactor coolant pumps remained operating instead of being tripped. As a basis for this reassessment, the reactor vendors were requested to consider, on a generic basis, all probable small break LOCA scenarios and the corresponding consequences for their plant designs. Our specific concerns were summarized in a letter(2) to Westinghouse (W), which identified 16 items related to the analyses of small break LOCAs, the consequences of failure or malfunction of primary components controlling system response to small break LOCAs, possible operator intervention with control system response when such failures or malfunctions occur, and the preparation of generic guidelines for emergency operational procedures to be followed in the event of small break LOCAs. The W Owners Group response to this request was summarized in a three-volume W report (WCAP-9600)(3) which addressed each of the 16 areas identified in our May 22, 1979 letter and included analyses for 35 different small break LOCA scenarios. These analyses apply to W plant designs generically, and were selected to the extent possible to produce bounding results for all operating W-designed plants for the accident scenarios in question.

The results of our review of this report are summarized in the next section. Further discussion of our findings on each of the principal areas of concern may be found in the sections that follow.

2. CONCLUSIONS AND RECOMMENDATIONS

2.1 Conclusion

The small break LOCA analysis methods used by W are satisfactory for the purpose of predicting trends in plant behavior following a small break LOCA. The results of the analyses can be used to develop improved emergency procedures and to train reactor operators. However, several individual analytical models have been identified in Section 4.2.1 as requiring either improvement or further confirmation. In addition, comparison of the total analysis method with available small break integral test data (Semiscale Test S-02-6) has indicated large uncertainties in the calculations. The analysis methods should be revised and verified before they can be considered for NRC approval under 10 CFR §50.46.

Recommendations

a. The analysis methods used by W for small break LOCA analysis for compliance with Appendix K should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT and Semiscale facilities.

b. Plant-specific calculations using the NRC-approved model, as described in a. above, for small breaks to show compliance with 10 CFR §50.46 should be submitted for NRC approval by all licensees.

2.2 Conclusion

Westinghouse has performed a sufficient spectrum of small break LOCA analyses to identify the anticipated system performance for breaks in this range. These analyses provide adequate bases for developing improved operator guidelines, and demonstrate that operator action and a combination of heat removal by the steam generators, high pressure injection system, and the break assure adequate core cooling. The required operator actions are the following: tripping the reactor coolant pumps shortly after occurrence of a LOCA, HPI termination in the event of a repressurization, and manual restoration of AFW or PORV opening in case of loss of all feedwater. Pump trip is required because W calculations show that, for a narrow range of small break sizes, the 10 CFR §50.46 limits on peak cladding temperature could be exceeded if the pumps are not tripped. According to W estimates, at least 10 minutes are available for the operator to perform this action. Our evaluation(4) of the W analysis indicates that the times available for the operator could be shorter than 10 minutes, and may even be as short as three minutes, indicating a need for automatic actuation.

System repressurization can occur for the smallest breaks following HPI actuation. When appropriate plant conditions are reached as specified in the guidelines for emergency procedures, operator termination of HPI is permissible to avoid filling the pressurizer with water and challenging the PORV.

If, in addition to the small break LOCA, all feedwater flow (both main feedwater and auxiliary feedwater) is lost, or if, for any reason natural circulation fails to take place, there will eventually (after at least 25 minutes, as discussed in Appendix VII) be no heat removal through the steam generators. In this case, operator action is required to restore feedwater flow or to open the pressurizer relief valves and block valves (if closed). According to W, in case of a loss of feedwater flow, either action will serve to depressurize the primary system so that sufficient safety injection flow can be established. If natural circulation fails, the operator must open the pressurizer PORVs. W indicated that approximately one hour is available for the operators to re-initiate feedwater flow. Opening of the relief valves must be accomplished within 40 minutes in order to keep the consequences of the event within acceptable limits. Our review of the W calculations revealed that W overestimated the relief valve flow rate used in the calculations and that the PORVs should be opened sooner. The importance of the flow rate through relief and safety valves is recognized in NUREG-0578,(1) Section 2.1.2, which requires that full-scale prototypical tests be performed by July 1981. Based on the previous discussion, we have concluded that a diverse decay heat removal path, independent of the steam generators, is desirable.

Recommendations

a. Tripping of the reactor coolant pumps in case of a LOCA is not an ideal solution. The licensees should consider other solutions to the small break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small break LOCA. The signals designated to initiate the pump trip should be carefully selected in order to differentiate between a small break LOCA and other events which do not require reactor coolant pump trip as discussed in NUREG-0623(4).

b. The W small break LOCA analyses relied on equipment which has not previously been characterized as part of the reactor protection system or part of the engineered safety features. The equipment used to provide reactor coolant pump trip, the pressurizer spray valves, the pressurizer relief valves, the pressurizer relief block valves, the equipment used to automatically actuate the pressurizer relief valves, and the equipment used to remotely control the pressurizer relief and block valves fall into this category. The reliability and redundancy of these systems should be reviewed and upgraded, if needed, to comply with the requirements of section 9 of NUREG-0585(5) regarding the interaction of non-safety and safety-grade systems.

c. Plant simulators used for operator training should offer, as a minimum, the following small break LOCA scenarios:

   (i) continuous depressurization,
   (ii) pressure stabilized at a value close to secondary system pressure,
   (iii) repressurization,
   (iv) stuck-open pressurizer relief valve(s), and
   (v) stuck-open letdown valve.

   Each of these cases should be simulated with the reactor coolant pumps running as well as with the reactor coolant pumps turned off. The first three events should be simulated for both cold and hot leg breaks. In addition to the usual single failures in the ECCS and feedwater systems, extended loss of feedwater (main and auxiliary), should be simulated in conjunction with the above events.

2.3 Conclusion

A number of concerns related to decay heat removal following a very small break LOCA, and other related items, were identified by Mr. C. Michelson of TVA (see Section 4.1 of this appendix). These concerns were identified for PWRs designed by Babcock & Wilcox and Combustion Engineering (6,7). W has reviewed these concerns and provided an analysis (3) of those items that relate to plants of their design. Postulated modes of two-phase flow natural circulation play an important role in the W analysis. The analysis provides an adequate assessment of these concerns. However, experimental results are not available to support the analytical predictions.

Recommendations

a. The various modes of two-phase flow natural circulation which are expected to play a significant role in plant response following a small break LOCA should be demonstrated experimentally. The results of the tests should be submitted for NRC review.

b. Appropriate means, including additional instrumentation if necessary, should be provided in the control room to facilitate verifying whether natural circulation has been established.

2.4 Conclusions

The record of PORV failures (to close) for all PWRs, 13 in approximately 200 reactor years, has demonstrated a potential cause of a small break LOCA. Ten PORV failures have occurred at B&W-designed plants, whereas two have been recorded for W-designed plants and one at a CE-designed plant. We have no evidence that this record is complete, nor do we have evidence to the contrary. In attempting to estimate the probability of small break LOCAs produced by PORV failure, we have evaluated data from the history of PORV openings at W-designed plants and the frequency of overpressure transients at these plants. We have also evaluated analytical predictions of PORV openings for these transients provided by W. We cannot conclude that the set of PORV opening data is sufficiently complete to permit accurate quantification of the probability of the valve failing to close.

Currently available results of analytical calculations with the LOFTRAN code cannot be used to quantify valve actuation probabilities because of the effect of certain conservative input, as discussed in Section 3.3 of this appendix.

Based on recommendations by the Lessons Learned Task Force (hereafter referred to as LLTF) the staff has required that redundant emergency power be provided for PORVs and associated block valves in all PWRs so that valves can open and close without the use of offsite power. In this way, the system can be used in the relief mode (both valves open), or the valves can be closed to prevent a small break LOCA under emergency conditions.

In addition, based on the LLTF recommendation (1), the staff has required that all PWRs, including those designed by W, be provided with either a direct indication of safety and relief valve position or with a reliable flow indication device to indicate open valve status. At present, most plants with NSSS designed by W have relief valve position indicators.

Finally, in its implementation of the LLTF recommendation (1), the staff has required that both safety and relief valves in PWRs be tested for functional performance capability. Some question arises because of the failure of the PORV at TMI-2 and that at the Beznau plant to close as to whether these valves have sufficient relief capability and can sustain the loads imposed during accidents in which two-phase slug flow or single-phase steam or water flow is relieved.

Based on our review, we have concluded that, in order to improve PORV reliability still further, licensees should design and install a control system which provides interaction between the PORV and its associated block valve to prevent a small break LOCA in the event of a failure of the PORV to close. One such design would cause the block valve to close after the PORV opens when the PORV reseat pressure has been reached, and the reactor coolant system pressure has decayed to a lower value (i.e., the block valve would remain open until its lower setpoint was reached). This system would be provided with an override so that pressure relief could be accommodated at lower pressures, as necessary. Justification would be required to assure the staff that failure of this system would not decrease overall safety by acting to intensify plant transients and accidents.

Recommendations

a. All PWR licensees should provide a system which uses the block valve to protect against a small break LOCA. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened, to relieve excess pressure. An override feature should be incorporated. Justification should be provided to assure that failure of this system would not decrease overall safety by intensifying plant transients and accidents.

b. Each licensee should perform a confirmatory test of the automatic block valve closure system installed in response to (a) above.

c. W should submit a report for staff review documenting the various actions which have been taken to decrease the probability of a small break LOCA caused by a stuck-open PORV and show how these actions constitute sufficient improvements in reactor safety.

d. Future failures of a PORV to close should be reported promptly to the NRC, consistent with the guidelines of NUREG-0610. Future challenges to the PORVs should be documented in the annual report.

2.5 Conclusion

One possible way to completely eliminate the risk associated with the failure of relief valves is to operate the plants with the block valves closed. This mode of operation, however, could result in an increase in the lift frequency of one safety valve. Information relative to the observed failure rate of safety valves has not been made available. Consequently, neither the desirability or acceptability of this mode of operation can be made at this time.

a. Safety valve failure rates based on past history of the W-designed plants should be included in the report specified in 2.4.c above.

b. Future failures of a safety valve to close should be reported promptly to the NRC. Future challenges to the safety valves should be documented in the annual report.

c. The staff's implementation of the Lessons Learned Task Force long-term recommendations (5) should pursue the interrelationship of safety and relief valves in its future study dedicated to safety and non-safety grade systems, Recommendation No. 9 of NUREG-0585.(5) This study should include an evaluation of the elimination of the PORV function.

2.6 Conclusion

A PORV supplied by Control Components, Inc. (CCI) used in the McGuire plant (owned by Duke Power) failed during hot functional testing. Because this valve is different from the Copes-Vulcan design, which comprises the operational data for W-designed plants, its failure mechanism and failure rate must be determined to be equivalent to that of the Copes-Vulcan valves, in order to include both in the same population. At present, a data base for operational failures for this valve does not exist.

Recommendation

Any plant using, or planning to use, this valve without modification should provide complete justification for such use in light of this failure. This matter should be addressed on a plant-by-plant basis.

The valve should be modified as recommended by the manufacturer and tested. Plants using this valve (modified or unmodified) should record each valve actuation and each valve failure. Failures must be reported to the NRC. The licensee must compare such failure with those of Copes-Vulcan valves with a view toward further modifications or replacement, if necessary.

2.7 Conclusion

The Pressure Integral Derivative (PID) controller, installed on at least one PORV in most W-designed plants, has caused spurious valve actuations. Westinghouse has recommended that the PID controller be modified in plants where it is used for PORV actuation by raising the interlock to the same valve as the PID control bistable, in order to estimate spurious openings.

Recommendation

The above described recommendation shall be completed by affected licensees.

2.8 Conclusion

Some W-designed operating plants want to modify the anticipatory trip (reactor trip upon turbine trip) so that it operates at power levels of 50% and above, instead of the present operational level of 10% above. Since many PORV openings have occurred at low power, this modification may increase the probability of a small break LOCA resulting from the failure of an opened PORV to close.

Recommendation

This proposed anticipatory trip modification should not be made until it has been shown on a plant-by-plant basis that the small break LOCA probability resulting from a stuck-open PORV is little affected by the modification.

3. EXPECTED FREQUENCY OF SMALL BREAK LOCAs

3.1 Introduction

The total expected frequency of a small break loss-of-coolant accident (LOCA) is the sum of the probability of a piping rupture and the probability of the failure of a valve which isolates the primary coolant system to close, if challenged, combined with the failure of the operator to isolate the stuck-open valve. The power-operated relief valve (PORV) is the most likely valve to fail in the open position. The median frequency of small reactor coolant system piping ruptures is estimated in Appendix V of WASH-1400 (8) to be 10^{-3} per reactor year for breaks between 1/2 and two inches equivalent diameter. The variation in this probability ranges from 10^{-2} to 10^{-4} per reactor year.* The probability of a small break LOCA from relief and safety valve failure was not given. However, the frequency of relief valve failure (to close) was stated to be 10^{-2} per challenge.

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*Steam generator tubes, whose diameters are also in this range have failed several times in a number of ways. To a first approximation, the tube failure rate appears indistinguishable, perhaps higher, than that for a stuck-open, unisolated PORV.

The incident at TMI-2 was initiated by a loss of feedwater transient, which caused a pressure increase in the primary system. This caused the subsequent opening of the PORV at the top of the pressurizer. The failure of the PORV to reseat when the primary system was depressurized below the valve setpoint pressure, together with the failure of the operator to isolate the PORV, produced the equivalent of a small break LOCA. We have evaluated the expected frequency of a stuck-open PORV by two methods in this study: (a) based on operating experience, and (b) based on analytical predictions of plant transient response. The former evaluation is based on a summary of operational data provided by W (9) and our evaluation of W-designed plant operating history contained in NUREG-0618(10), while the latter evaluation is based on analyses of anticipated transients provided in WCAP-9600 (3).

3.2 Data from Operating Reactors

In the past 200 reactor years of pressurized water reactor (PWR) operation, so far 250 instances of PORV openings have been recorded. We have no evidence that this record is complete. In 13 of these instances, the PORV failed to close. Most of the recorded openings (162) and valve failures (10) have occurred at B&W-designed plants.

Only one instance of a safety valve opening has been recorded. This occurred at a B&W-designed plant. The valve reseated properly after opening. This record may also not be complete.

There have been two recorded failures of PORVs in nuclear power plants with nuclear steam supply systems (NSSS) designed by W. One is the failure during power operation of a Copes-Vulcan valve (typical of the PORV design used in most plants with NSSS designed by W) at the Beznau reactor in Switzerland. The other failure occurred when a PORV supplied by Control Components, Inc., (CCI), which is being used in the McGuire plant owned by Duke Power, failed during hot functional testing. This plant has not yet received an operating license.

The CCI valve is used at only one plant and the design must be modified to prevent future failures. For these reasons, perhaps this PORV failure should not be included in the data used to judge future operation of W-designed plants. On the other hand, we have no assurance that the two recorded valve failures represent a complete list.

We have therefore recommended that W document an evaluation of past experience with PORVs in W-designed plants in the context of whether the deterministic alterations to PORVs and block valves are sufficient remedies.

Using the fact that W-designed plants have accumulated a total of approximately 150 reactor years of operation and the recorded number of PORV failures mentioned above (two), a PORV failure (to close) rate for W-designed plants of 0.0133 per reactor-year is calculated. When the industry-wide recorded failure data are used, a failure rate of 0.065 per reactor-year is obtained. Before a comparison can be made with the expected frequency of a small break LOCA due to pipe rupture, these failure rates must be combined with the probability that a PORV fails in the open position. However, as mentioned previously, we have no assurance that the list of recorded PORV failures is complete.

The fail-open rate for relief valves is given in WASH-1400 (8) as 10^{-2} per demand. If the challenge rate for PORVs at W-designed plants could be established with certainty, a more reliable means for estimating the rate of PORV failure to close per reactor-year would result.

In a letter to the NRC (9), W documented 43 PORV openings in 141 years of reactor operation. This list was stated by W to be incomplete. In discussions with licensees with W-designed plants, a total of 60 PORV openings was reported to the NRC staff (see Table VIII-1). We do not know whether or not this list is complete. The procedures used by the owners to assure completeness were not documented. Considering that W-designed plants are supplied with at least two PORVs, each of which opens if the set-point pressure is reached, and using the W data in Reference 9 and the PORV fail open probability given in WASH-1400, (8) the expected fail-open rate per reactor-year would be 6×10^{-3}, which is within the range of probabilities for small pipe ruptures stated in WASH-1400.(8) Similarly the fail-open rate calculated using the data reported to the NRC by the licensees with W-designed plants is approximately 8×10^{-3} per reactor- year.

Although PORV openings are not routinely recorded, except on reactor pressure strip charts, reactor trips are reported to the NRC. In one year (1978) W-designed operating plants recorded 150 reactor trips. Of these, 76 were produced by transients which might have caused system overpressure. For the 23.4 reactor years of operation for W-designed plants in 1978, the rate of possible overpressure events is 3.24 per reactor-year. Not all these events would result in the PORV opening, since the severity of overpressurization transients is reduced in most instances by anticipatory reactor trips. These include reactor trips on turbine trip, high and low steam generator level, and primary and secondary system temperature mismatch. Of the 76 possible overpressurization events which occurred at W-designed plants in 1978, turbine and feedwater transients produced most of the reactor trips. Turbine trips produced approximately 20% of these events and feedwater transients produced 55%. The remainder of the events were produced by loss of heat sink events other than a turbine trip. The next section discusses the analytical predictions performed by W for these transients to determine which overpressure events will cause the PORV to open.

Some further consideration must be given to the role of the block valve in case of failure of the PORV to close, and to the effect on a small break LOCA if the PORV opened, stuck open, and the block valve were closed for some reason. In most recorded cases of PORV failure to close, actions of the operator in closing the block valve limited the event. In the case of TMI-2, the operator failed to close the block valve for two hours after the PORV failed to close. Plant operating procedures implemented subsequent to the TMI-2 accident have placed emphasis on closing the block valve in the event of a LOCA. Most W-designed operating plants have both a thermocouple located in the discharge region of the PORV and direct indication of valve position (the staff has already taken steps as described in NUREG-0578 (1) to upgrade PORV and block valve performance). Operators in those plants with NSSS designed by W should respond promptly to indications of an opened PORV. For this reason, the likelihood of a stuck-open PORV and an open block valve resulting in a LOCA would be less than the probability that a PORV will fail to automatically close.

Neither the block valve operator nor its indication is environmentally qualified, safety-grade, or single failure-proof. The PORVs are not tested by the manufacturers, but are tested during the plant's pre-operational phase by the licensee. We do not know whether the block valves are also tested during this phase. The block valves, however, have functioned as required to isolate the PORV, once the stuck-open PORV was detected by the operator. Currently, plants may continue to operate even if the block valves are inoperative. There is no limit on plant operation in plant technical specifications specifying how long the block valves may remain in this condition. For the present, we intend to require that the block valve be automatically closed once the PORV has opened and the reactor coolant system pressure has decreased below the PORV reseat pressure. We will also require that this capability be demonstrated by test at each plant.

3.3 Evaluation of Anticipated Transients

W has performed analyses of anticipated transients using the LOFTRAN computer code to identify those transients which might cause the PORV to open. These analyses were performed for a typical four-loop plant.

The LOFTRAN digital computer code is described in Topical Report WCAP-7907, "LOFTRAN Code Description," November 1978. Supplementary information is contained in letters NS-TMA-1802, dated May 26, 1978, and NS-TMA-1824, dated June 16, 1978. The LOFTRAN code is an analytical model of the primary and secondary system of a W-designed pressurizer water reactor. Time-dependent neutron kinetics equations are solved for the fuel pins in conjunction with the solution of thermal-hydraulic equations for the reactor coolant system.

The LOFTRAN computer program contains several models that are desirable for the evaluation of anticipated transients. These include a non-equilibirum pressurizer model, and a detailed primary-to-secondary system heat demand model. The non-equilibrium pressurizer model calculates higher primary system pressures for insurges into the pressurizer than the WFLASH code, which utilizes an equilibirum pressurizer model. The steam generator model calculates the spatial temperature gradient in the primary side of the steam generator U-tubes, and provides a more detailed representation of primary-to-secondary system heat transfer than WFLASH. Pressurizer sprays, heaters, PORVs, and safety valves are modeled in LOFTRAN. The program can represent up to four primary system loops. The fluid in these loops can be either liquid or saturated two-phase. Program limitations are that the primary system fluid in the loops must be homogeneous and undirectional, although reverse flow is permitted in an entire loop.

The LOFTRAN computer program has been reviewed and conditionally approved (11) by the NRC staff for use in the analysis of anticipated transients without scram (ATWS) and is currently under review for use in the analysis of other transients. Predictions by LOFTRAN have been found to compare favorably with plant data for pump trip tests, load increase, and load rejection transients, including a 100% load rejection transient. No data comparisons with LOFTRAN have been presented for loss of feedwater transients, loss of offsite power or steam generator isolation transients.

Using the LOFTRAN code, analyses of anticipated transients were performed by W including: (1) loss of offsite power, (2) loss of normal feedwater, (3) loss of load, and (4) contro rod bank withdrawal at power. The transients were analyzed utilizing input assumptions categorized as "worst expected case" and "FSAR". The differences in input are compared in Table VIII-1. Peak reactor system pressures predicted by these analyses are also presented in Table VIII-2. For the "worst expected case," core physics assumptions were used that would be conservative for 95% of core life. Nominal values were used for protective system setpoints and for plant initial conditions. The control systems (e.g., pressurizer pressure control system) were assumed to operate as designed. The "FSAR" cases utilized conservative input values designed to bound actual plant conditions A constant PORV setpoint of 2350 psig was assumed.

As indicated by Table VIII-2, the PORV was calculated to lift for loss of feedwater and loss of load, but not for turbine trip accompanied by an anticipatory scram. The PORV was also calculated to lift for the rod bank withdrawal at power transient. LOFTRAN predicted that the pressurizer liquid level would remain below the PORV exit elevation, so that steam, not liquid, was calculated to flow through the PORVs and safety valves when they were calculated to flow through the PORVs and safety valves when they were opened. The amount of steam release reduced system pressure sufficiently so that the PORV closed before any liquid could be discharged. We have reviewed the frequency of overpressure transients at W-designed plants, and conclude that the occurrence of rod withdrawal transients is highly unlikely in comparison with that for loss of load and loss of feedwater transients.

Safety valves were calculated to lift for loss of load transients without anticipatory reactor trips, if the PORVs were assumed to fail closed. These analyses indicate that operation with the block valves upstream of the PORVs in the closed position may result in an increased number of safety valve challenges.

The comparison of W plant data in Reference 9 and the LOFTRAN results shown in Table VIII-2 differ in some respects. LOFTRAN predicts that the PORVs remain closed following a turbine trip, whereas W[9] has tabulated two instances of PORVs opening on turbine trip. The details of these two instances (how the turbine tripped, the turbine bypass capacity, etc.) have not been examined. An additional two instances have been tabulated of PORVs opening following a loss of offsite power, whereas LOFTRAN predicts that the valve will remain closed. Loss of feedwater is predicted to cause the valve to open and WASH-1400 [8] estimates that three such events will occur per reactor year. As discussed in Reference 9, W has tabulated no instances of PORVs opening on loss of feedwater in 141 total years of operating experience for W-designed plants. Some of the above differences may result from the conservative input used in the LOFTRAN analyses whose results are shown in Table VIII-2. The "worst expected case" analyses used core physics parameters which would be representative of only 5% of the plant cycle.

The assumptions used for the PORV pressure controls may also explain the differences between the plant data and the LOFTRAN predictions. The LOFTRAN predictions assumed a constant PORV setpoint of 2350 psia, whereas most Westinghouse plants have a rate-dependent PORV setpoint. It is apparent that the LOFTRAN calculations cannot be used in an attempt to verify whether the observed number of plant transients exceeds the expected design value.

The W-designed operating plants differ in number of loops, power level, steam generator design, and PORV capacity. However, the PORVs and safety valves have been designed to the same criteria for the individual plants. Hence the analytical results discussed above should be representative of all the W-designed operating plants.

3.4 Plant Modifications to Reduce the Possibility of Small Break LOCA Resulting from a Stuck-Open PORV

3.4.1 Discussion

Subsequent to the TMI-2 accident, the PORV pressure setpoint at B&W-designed operating plants was increased and the high pressure reactor trip setpoint was decreased. Anticipatory reactor trips were installed for turbine trip and new B&W setpoints as well as those of CE and W are given in Table VIII-3. No incidents of PORV openings have occurred at B&W-designed plants since these modifications were installed, although 12 transient events, which would have previously caused the PORV to open, have been reported.(12)

Modification of setpoints in a similar manner for W plants could reduce the probability of a small LOCA by reducing the number of PORV challenges. However, W reported that, if modifications were made to prevent PORV opening with sufficient margin to prevent safety valve actuation, spurious reactor trips would occur; e.g., for a main steam isolation valve (MSIV) closure event, with the PORV setpoint at 2400 psig (100 psi below the safety valve setpoint) and the high pressure reactor trip 150 psi below that, the high pressure trip would fall into the normal pressure control range.

Westinghouse also noted that, even if this method had been used successfully, it would not have eliminated all the past PORV actuations. That this is true can be seen from Table VIII-4.

Although the modification of the PORV and high pressure reactor trip setpoints was considered during our review, it was not adopted for implementation on W-designed operating plants. The bases for this decision are discussed later in this section.

The proportional integral derivative control system (PID) utilized for at least one PORV in most W-designed plants, has caused spurious openings, because the pressure level at which the PID controllers were permitted to operate the PORVs was considerably below the operating pressure. Westinghouse has recommended that the PID controllers be replaced so that all PORVs will open at a constant pressure. This has already been done at the Cook, Ginna, and Trojan plants. We recommend that this change be made at all W-designed operating plants.

After the PID controllers have been replaced, W-designed operating plants will operate with two or three PORVs, which are all set at the same pressure (2350 psig). For any transient where the reactor system pressure reaches this value, more than one valve will open. Some transients require the relieving capacity of only one valve so that, if the pressure setpoints were staggered, only one valve would open.

One way to prevent PORVs from failing to close is to prevent them from opening--by securing them in the closed position, together with a secured block valve. As can be seen from Table VIII-3, this increases the possibility of safety valve actuation for loss of load transients without anticipatory trips. However, unnecessary openings of safety valves (which are not permitted to have block valves in the discharge line) are not desirable, since a failure of a safety valve (which is non-isolable) to reseat inevitably results in a LOCA. By design, safety valve openings are reduced by utilization of relief valves (PORVs).

Actuation of the safety valve instead of the PORV would be preferable if the reliability of the PORV were found to be significantly less than that of safety valves. However, no data have been made available to show whether safety valves are more or less reliable than relief valves for this failure mode. Elimination of PORVs should be considered if the probability of a small break LOCA resulting from a stuck-open PORV is found to be significantly larger than the probability of a small break LOCA resulting directly from a pipe break. This can be examined after receipt of the recommended report from W regarding the efficacy of PORV and block valve upgrading.

The TMI-2 Lessons Learned Task Force (hereinafter referred to as LLTF) has recognized the need for evaluating the relationship between safety and nonsafety grade systems as described in Recommendation No. 9 of NUREG-0585(5). This evaluation must include the relation between safety and relief valves in order to assure necessary plant safety. As a part of that long-term evaluation, the possible elimination of PORVs should be examined.

Some licensees wish to modify their plant protection system so that the reactor trip resulting from turbine trip in the power range above 10% is changed to require reactor trip only above a power level of 50%. From a cursory examination, it appears that this change will have a slight effect on plant safety, because of the limited time the plants operate in this power range. However, further study is required to confirm that this change will not significantly increase the probability of PORV failure to close. This study will have to include an examination of the fraction of total operating time spent in this power range. It will also have to determine that the likelihood of PORV actuation and failure is no greater in this region (10-50% power level) than in the region above 50% of full power. The anticipatory trip modification should not be made until it is shown, either generically or on a plant-by-plant basis, that the small break LOCA probability resulting from an open, unisolated PORV is little affected by this proposed change since many PORV openings occur at low power.

Based on the LLTF short-term recommendations (1) the staff has required that redundant emergency power be provided for PORVs and associated block valves in all PORVs so that valves can open and close without the use of offsite power. In this way, the system can be used in the relief mode (both valves open), or the valves can be closed to prevent a small break LOCA under emergency conditions.

In addition, based on the LLTF short-term recommendations (1), the staff has required that all PWRs including those designed by W, be provided with either a direct indication of safety and relief valve position or with a reliable flow indication device to indicate open valve status. At present, most plants with an NSSS designed by W have relief valve position indicators.

Finally, in its implementation of the LLTF recommendations, the staff has required that both safety and relief valves in PWRs be tested for functional performance capability. Some question arises, because of the failures to close of the PORV at TMI-2 and that at the Beznau plant in Switzerland, as to whether these valves have sufficient relief capability and can sustain the loads imposed during accidents in which two-phase slug flow or single-phase steam or water flow is relieved.

Based on our review, we recommend that, in order to improve PORV reliability still further, licensees should design and install a control system which provides interaction between the PORV and its associated block valve to prevent a small break LOCA in the event of a failure of the PORV to close. One such design would cause the block valve to close after the PORV opens, when the PORV reseat pressure has been reached, and the reactor coolant system pressure has decayed to a lower value (i.e., the block valve would remain open until its lower setpoint was reached). This system would be provided with an override so that pressure relief could be accommodated at lower pressures, as necessary. Justification would be required to assure the staff that failure of this system would not decrease overall safety by acting to intensify plant transients and accidents.

3.4.2 Additional Considerations

One member of the Bulletins & Orders Task Force proposed that the task force adopt the following recommendations to reduce the likelihood of a stuck-open PORV.

(a) Frequent overpressure transients should not result in opening of the relief valves. Licensees with W-designed plants should:

   (i) install an anticipatory reactor trip on turbine trip, if it is not already present,

   (ii) show that overpressure feedwater transients challenge the relief valves only in exceptional cases (less than 5 percent of transients). If this point cannot be supported by the 150 years of operating experience currently available, appropriate design changes should be introduced. These changes should include a change in the relative position of the relief valve and high pressure reactor trip setpoints and the installation of additional anticipatory reactor trips.

   (iii) Licensees must comply with this requirement by March 1, 1980.

(b) Licensees with W-designed plants have reported approximately 300 feedwater transients to date. The peak reactor system pressure reached during these transients and possible indication of relief valve opening during these transients should be reported to the NRC before March 1, 1980.

(c) All future relief valve challenges should be recorded and reported to the NRC.

(d) Each licensee should be required to submit a report on safety valve challenge rate and safety valve failure rate, based on the operating history of the plant. These reports should be submitted to the NRC by February 1, 1980.

(e) All future safety valve challenges should be recorded and reported to the NRC.

Although the Bulletins and Orders Task Force (B&OTF) gave the above-described proposal serious consideration, we decided not to adopt it in its entirety for the following reasons:

(a) As a result of the staff's implementation of the recommendations specified in NUREG-0578,(1) it is required that PORVs and associated block valves be powered off emergency buses, that PORVs be performance-tested under appropriately challenging conditions, and direct position indication be provided in the control room for PORVs.

(b) Through the work of the B&OTF, plant procedures have been upgraded to account for a stuck-open PORV.

(c) The PID control system, which is installed on at least one PORV in most W-designed plants, is being eliminated. This control system resulted in a considerable number of relief valve challenges.

(d) Safety classifications and qualifications are being upgraded through the implementation of Recommendation No. 9 of NUREG-0585 (5).

(e) We have recommended a new design feature for all PWRs which incorporates automatic closure of the block valve on low pressure that will isolate any stuck-open PORV without the need for operator detection and action.

(f) We have recommended that W prepare a report for staff review describing the various actions identified in the preceding section of this report which have been taken to decrease the probability of a small break LOCA caused by a stuck-open PORV and show how these actions constitute sufficient improvements in reactor safety.

3.5 Conclusions and Recommendations

3.5.1 Conclusions

The record of PORV failures to close, 13 in approximately 200 reactors years, has demonstrated a potential cause of a small break LOCA. Ten PORV failures have occurred at B&W-designed plants, whereas two have been recorded for W-designed plants and one at a CE-designed plant. We have no evidence that this record is complete, nor do we have evidence to the contrary. In attempting to estimate the probability of small break LOCAs produced by PORV failure, we have evaluated data from the history of PORV openings at W-designed plants and the frequency of overpressure transients at these plants. We have also evaluated analytical predictions of PORV openings for these transients provided by Westinghouse.

We cannot conclude that the set of PORV opening data is sufficiently complete to permit accurate quantification of the probability of the valve failing to close.

Currently available results of analytical calculations with the LOFTRAN code cannot be used to quantify valve actuation probabilities because of the effect of certain conservative input, as discussed above.

Based on recommendations by the Lessons Learned Task Force (1), the staff has required that redundant emergency power be provided for PORVs and associated block valves in all PWRs so that valves can open and close without the use of offsite power. In this way, the system can be used in the relief mode (both valves open), or the valves can be closed to prevent a small break LOCA under emergency conditions.

In addition, based on the LLTF recommendations (1), the staff has required that all PWRs, including those designed by W, be provided with either a direct indication of safety and relief valve position or with a reliable flow indication device to indicate open valve status. At present, most plants with an NSSS designed by W have relief valve position indicators.

Finally, in its implementation of the LLTF recommendations (1), the staff has required that both safety and relief valves in PWRs be tested for functional performance capabilit Some question arises, because of the failure of the PORV at TMI-2 and that at the Beznau plant to close, as to whether these valves have sufficient relief capability and can sustain the loads imposed during accidents in which two-phase slug flow or single-phase steam or water flow is relieved.

Based on our review, we have concluded that, in order to improve PORV reliability still further, licensees should design and install control system which provides interaction between the PORV and its associated block valve to prevent a small break LOCA in the event of a failure of the PORV to close. One such design would cause the block valve to close after the PORV opens when the PORV reseat pressure has been reached, and the reactor coolant system pressure has decayed to a lower value (i.e., the block valve would remain open until its lower setpoint was reached). This system would be provided with an override so that pressure relief could be accommodated at lower pressures, as necessary. Justification would be required to assure the staff that failure of this system would not decrease overall safety by acting to intensify plant transients and accidents.

Recommendations

(a) All PWR licensees should provide a system which uses the block valve to protect against a small break LOCA. This system will cause the block valve to close automatically when the reactor coolant system pressure decays after the PORV has opened, to relieve excess pressure. An override feature should be incorporated. Justification should be provided to assure that failure of this system would not decrease overall safety by intensifying plant transients and accidents.

(b) Each licensee should perform a confirmatory test of the automtic block valve closure system installed in response to (a) above.

(c) Westinghouse should prepare a report documenting the various actions which have been taken to decrease the probability of a small break LOCA caused by a stuck-open PORV and show how these actions constitute sufficient improvements in reactor safety.

(d) Future failures of a PORV to dose should be reported promptly to the NRC. Future challenges to the PORVs should be documented in the annual report.

3.5.2 Conclusion

One possible way to completely eliminate the risk associated with the failure of relief valves is to operate the plants with the block valves closed. This mode of operation, however, could result in an increase in the lift frequency of one safety valve. Information relative to the observed failure rate of safety valves has not been made available. Consequently, neither the desirability nor acceptability of this mode of operation can be made at this time.

Recommendations

(a) Safety valve failure rates based on past history of the W-designed plants should be included in the report specified in 3.5.1.c above.

(b) Future failures of safety valves to close should be reported promptly to the NRC. Future challenges to the safety valves should be documented in the annual report.

(e) The staff's implementation of the Lessons Learned Task Force long-term recommendations (5) should pursue the interrelationship of safety and relief valves in the future study dedicated to safety and non-safety grade systems, Recommendation No. 9 of NUREG-0585.(5) This study should include an evaluation of the elimination of the PORV function.

3.5.3 Conclusion

A PORV supplied by Control Components, Inc. (CCI) used in the McGuire plant (owned by Duke Power) failed during hot functional testing. Because this valve is different from the Copes-Vulcan design, which comprises the operational data for W-designed plants, its failure mechanism and failure rate must be determined to be equitable with that of the Copes-Vulcan valves, in order to include both in the same population. At present, a data base for operational failures for this valve does not exist.

Recommendation

Any plant using, or planning to use, this valve without modification should provide complete justification for such use in light of this failure. This matter should be addressed on a plant-by-plant basis

The value should be modified, as recommended by the manufacturer, and tested. Plants using this valve (modified or unmodified) should record each valve actuation and each valve failure. Failures must be reported to the NRC. The licensee must compare such failure with those of Copes-Vulcan valves with a view toward further modification or replacement, if necessary.

TABLE VIII-1

TRANSIENT ANALYSIS ASSUMPTIONS:

INITIAL CONDITIONS	Case	FSAR
- Power	Rated	E_sF + Calorimetric
- Pressure	P_{nom}	$P_{nom} \pm 30$
- Temperature	T_{avgnom}	$T_{avgnom} \pm 4$
- Steam Generator Level	T_{nom}	$L_{nom} \pm 5\%$ Level Error ± 10% Mass Uncertainty Error
- Rod Drop Time	BE	TD Flow and Uncertainties
- RCS Flow	BE Flow	TFD
- Decay Heat	ANS	Appendix K
- Reactivity Coefficients	Conservative for 95% of core life	Bounding
- Trip Setpoints	PLS	PLS ± Instrument Errors
- Auxiliary Feedwater Flow	Rated	Worst Single Active Failure
- Safety Injection Flow	Rated flow rates with No assumed failures	Minimum Safeguards Flow

TABLE VIII-2

SUMMARY OF PEAK PRESSURIZER PRESSURES

ASSUMPTIONS	WORST(1) EXPECTED CASE	WORST EXPECTED CASE	WORST EXPECTED CASE	FSAR	FSAR
Control systems available	YES	NO	YES	NO	NO
PORV available	YES	YES	NO	YES	NO
1. Loss of Offsite Power	2291	2272	2291	2370*	2529**
2. Loss of Normal Feedwater	2352*	2354*	2388*	2371*	2529**
3. Loss of Load					
a. Turbine Trip	2262	2262	2258	2209	23-3
b. Turbine Throttle Closure	2356*	2369*	2508**	2462*	2537**
c. MSIV Closure	2437*	2489*	2530**	2499*	2542**
4. Rad Bank Withdrawal at Power	2354*	2353*	2451*	2435*	2535*

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* PORV opens (set pressure 2350 psia)

** Safety valve opens (set pressure 2500 psia)

(1) See Table VIII-2 for definition

TABLE VIII-3

TYPICAL SETPOINTS FOR RELIEF VALVE, SAFETY VALVES AND OVERPRESSURE TRIP

Setpoints (psia)	B&W Prior to 3/79	B&W Present	CE	Westinghouse
Operating Pressure	2155	2155	2250	2250
Relief Valve	2255	2450	2400	2350
Overpressure Trip	2355	2300	2400	2400
Safety Valve	2500	2500	2500	2500

TABLE VIII-4
SUMMARY OF PORV OPENING DATA FOR WESTINGHOUSE-DESIGNED OPERATING PLANTS

Instrument or Technical Error	9
Intentional Opening for Test	16
Intentional Opening for Pressure Control	2
Cold Shutdown Water Solid	5
Cause Unknown	1
Transient-Automatic Response	27
Total	60

4. EXPECTED CONSEQUENCES OF SMALL LOCAs

4.1 TVA Small Break LOCA Concerns as Applied to Westinghouse-Designed Plants

4.1.1 Background

In a meeting between Combustion Engineering (CE) and the Tennessee Valley Authority (TVA) on May 27-28, 1975, regarding emergency feedwater, TVA expressed concerns about a class of small break LOCAs whose depressurization rates are slower than those analyzed in the CE System-80 Standard Safety Analysis Report (CESSAR)(13). The basis for the concerns was that, for this class of very small break LOCAs, makeup rates from the high pressure injection pumps may be inadequate for core cooling, leading to uncovering of the core with unacceptable peak cladding temperatures.

Subsequent to this meeting, Mr. C. Michelson of TVA drafted a report (6) based on "an ongoing qualitative consideration of this concern..." in which a number of questions were expressed on decay heat removal during post-small break LOCA recovery. More recently, in a letter (7) to the Babcock and Wilcox Company (B&W) dated April 26, 1978, TVA documented concerns regarding the ability of the B&W 205 Fuel Assembly (FA) plants to adequately remove decay heat during small break LOCA recovery. Of the six concerns identified for B&W 205-FA plants, two were specific to B&W designs, and four were generic to PWRs.

These four generic concerns and the concerns identified in Reference 6 have been addressed by W in Reference 3 regarding applicability to W-designed plants. In some instances, the specific concerns of Michelson served as a basis from which a broader-based staff concern evolved. We have reviewed the W evaluation of both sets of Michelson's concerns, (6,7) and the broader-based concerns subsequently identified by the staff. The results of this review and the conclusions are provided in the following sections.

4.1.2 Pressurizer Level as a Correct Indication of Water Level in the Core (Generic, from Reference 7)

Concern was expressed by TVA that, during a small break LOCA, the loop seal in the pressurizer surge line would prevent the pressurizer from draining.

Pressurizer level behavior during a small break LOCA is dependent upon the location of the break. If the break is not in the pressurizer, then the indicated level is expected to provide a reasonably accurate measure of primary system inventory. This is because the pressurizer will remain the high pressure point in the system until the core fluid reaches saturation. While it is the high pressure point, negligible voids should form in the rest of the primary system. The mass lost out the break will cause the indicated pressure level to be lost (and therefore of no more use to the operator) before there is any significant void buildup in the primary system.

For small breaks in the pressurizer vapor space (in particular, a stuck-open relief or safety valve), pressurizer level will give the operator an erroneous indication of primary system coolant inventory. This is because the pressurizer becomes the primary system low pressure point. Significant voiding can now take place in the reactor vessel, while the pressurizer instrumentation will indicate a high level because of the liquid insurge to the pressurizer from the primary system. It was this phenomenon which misled the operators at TMI-2 into thinking that the primary system was filled with liquid and therefore, they should shut off the HPI pumps.

We agree that pressurizer level is not an accurate measure of system coolant inventory under small break LOCA conditions. It is our conclusion that, by itself, pressurizer level should not be used to automatically actuate or terminate the Emergency Core Cooling System (ECCS). This was previously expressed in IE Bulletin 79-05A, which specified HPI termination criteria acceptable to the staff.

Westinghouse has also recognized this concern, and has recently modified the ECCS actuation signals for their plant designs, such that the ECCS will be activated only on low pressurizer pressure, rather than on coincidence of pressurizer pressure and level as was previously done.

The HPI termination criteria originally proposed by the W Operating Plants Owners' Group in WCAP-9600(3) stated that HPI can be terminated provided (a) reactor coolant system (RCS) pressure is above 2000 psig and increasing, (b) pressurizer water level is greater than 50% of span, and (c) water level is present in at least one steam generator. Item (b) of these criteria relies upon pressurizer water level. Moreover, these criteria describe characteristics that could apply to a loss of heat sink event. System repressurization to the PORV or safety valve setpoint would be greater than 2000 psig, and liquid inflow into the pressurizer could result in an indicated level in excess of 50% of span. We believe that the pressurizer level is a valid indication of system inventory only when it can be demonstrated that no voids exist in the primary coolant. This can most readily be accomplished by showing that an acceptable margin of subcooling exists in the primary coolant. Therefore, we required that these termination criteria be revised to (a) only allow reliance on pressurizer level when the operator can confirm that a sufficient subcooling margin exists in the primary loops and (b) insure HPI termination is precluded in the event of a loss of heat sink that exhibits the necessary characteristics which would permit HPI termination. The W Owners' Group revised the small break LOCA guidelines for operational procedures regarding HPI termination criteria to meet these requirements. Our approval of the guidelines was given in References 14, 15, and 16.

4.1.3 Small Break Isolation

The isolation of small breaks and subsequent repressurization of the RCS was expressed as a concern by TVA. This involved postulating a small break in a location that could be isolated by the operator (e.g., letdown line, PORV relief line) at some time after the break occurred (this was the situation that occurred at TMI-2, in which the stuck-open relief valve was isolated a few hours into the accident). The basis for this concern was that this event scenario had not been explicitly analyzed. In particular, break isolation without feedwater could cause repressurization and shut off the HPI flow.

In response to this concern, W̲ evaluated both the ability of the operator to isolate breaks and the consequences of isolating a break at the most inopportune time during the transient. Based on consideration of the accident sequence, W̲ determined that break isolation would cause some system repressurization. The potentially most inopportune time to isolate a break was therefore considered to be at the time of minimum system liquid inventory. The basis for this was that the void compression due to repressurization would have the greatest likelihood of producing or increasing the amount of uncovering of the core. Moreover, with or without degraded or no secondary system heat removal, repressurization could decrease or stop the HPI flow, and thus increase the potential for extending the inventory recovery time and exceeding the cladding temperature limits of 10 CFR §50.46.

Westinghouse reported that there are piping connections within the primary system coolant boundary that could be isolated if broken. These include the letdown line, the pressurizer PORV valves, and the pressurizer spray line. Two cases were analyzed by W̲. The first was a case in which three PORVs in the pressurizer were assumed to stick open, and were then isolated at the time of minimum system inventory. The second was a cold leg break simulating a break in the letdown piping, which was also isolated at the time of minimum system inventory. The pressurizer spray line was not analyzed, since it is automatically isolated on low pressurizer pressure.

4.1.3.1 Case 1: PORV Isolation with and without Auxiliary Feedwater

The break in the pressurizer corresponded to three stuck-open PORVs, with an equivalent break area of 0.034 ft^2. The time of minimum system inventory (when HPI flow equals break flow) was calculated to occur at 4800 seconds. At that time the three PORVs were assumed to be isolated by the operator. In addition, it was assumed that auxiliary feedwater was not available.

For this case, PORV isolation was of significant benefit, since when the PORVs were isolated, the pressurizer liquid inventory was calculated to drain back into the core, greatly increasing the core coolant inventory. Once the break was isolated, the system began to pressurize until primary to secondary heat transfer was established. Heat removal by the steam generators was still calculated to occur because the steam generator secondary inventory had not yet been calculated to boil off completely. System refilling was also enhanced by isolating the break, since the mass previously being lost out of the system was now being retained, and this effect more than offset the decreased HPI flow due to the slight repressurization. Natural circulation was calculated to be established at 7600 seconds. If the operators could not reestablish auxiliary feedwater flow, the steam generators would dry out, and the system would repressurize. In addition, failure to establish natural circulation (e.g., because of postulated loop blockage) would also cause the system to repressurize. Under these conditions, the operators would then be relied upon to reopen the PORVs, and decay heat removal would continue with a "feed and bleed" mode of operation utilizing the HPI pumps and the PORVs.

It was also considered whether a smaller break (i.e., one or two stuck-open valves) might produce more limiting results, since the time of minimum inventory would be later than the reference case, and the steam generator secondary inventory might be depleted at this later time. Thus, if core uncovering was calculated at the time of isolation, rapid repressurization due to lack of secondary heat transfer could quickly stop HPI flow before the core was recovered.

For the one and two stuck-open valve cases, while the time to minimum inventory is increased, no uncovering of the core is predicted at this time of minimum inventory. Therefore, repressurization at the time of break isolation due to lack of secondary heat transfer will not produce uncovering of the core and the results will be bounded by the three stuck-open valve case which was analyzed.

An additional case was examined by W which included the restoration of auxiliary feedwater flow at the same time that the PORVs were isolated. This case showed that, once the break was isolated at 4800 seconds and the pressurizer drained, a continuous system depressurization was predicted, with natural circulation being established by 7000 seconds, and the accumulator setpoint pressure reached by 8000 seconds.

4.1.3.2 Case 2: Letdown Line Break and Isolation

An analysis was performed of a two-inch break in the cold leg, which represented a break in the letdown line. Previous analyses showed that the time of minimum system inventory was 1900 seconds, and this time was chosen to isolate the break. This calculation showed that no additional uncovering of the core was predicted, due to isolation of the break, and that the system refilled at a faster rate, since the inventory loss out the break was stopped.

The time of minimum system inventory does not coincide with the time of maximum uncovering of the core. Had the break been isolated at the time of maximum core uncovering, the level of uncovering would have been slightly increased by a few inches, but inventory recovery would have started sooner.

4.1.3.3 Pressurizer Spray Line Break

Westinghouse did not analyze the case of a break in the pressurizer spray line, since it is automatically isolated on low pressurizer pressure. However, the controller for this valve is not a safety-grade system, nor can this line be isolated if the isolation valve is postulated to fail (single active failure). Since this scenario has not been analyzed and because it would involve fluid discharge simultaneously from the pressurizer and cold leg, W should provide an analysis assuming of this break location.

4.1.4 Pressure Boundary Damage Due to Bubble Collapse (Generic, from Reference 7)

The TVA letter discussed the possibility of damage to system components due to water hammer effects from condensing steam. In particular, the TVA concern focused on the injection of cool water by the high pressure injection system into a steam-filled colc leg pipe, as well as the bubbling of steam through subcooled liquid.

4.1.4.1 Steam Bubbles in Subcooled Liquid

Westinghouse has reported that they have not identified any situations in which steam bubbles will enter a region of subcooled liquid and collapse. We rule out the possibility of such a situation existing during a small break LOCA, because computer models currently used to calculate plant response to small break LOCAs do not account for nonequilibrium conditions. If the bubbling of saturated steam through subcooled water did occur, pressure pulses would be nondirectional, and a system containing steam bubbles would be hydraulically "soft," thereby attenuating pressure pulses prior to contact with structural boundaries. It is expected that these loadings would be bounded by the more severe case of cold water injection into steam discussed in Section 4.1.4.2 below. In our judgment, the loadings caused by steam bubbles will not exceed those associated with the design basis large break LOCA.

4.1.4.2 Cold Water Injection into a Steam-Filled Pipe

Westinghouse has estimated that the loads which would result from the injection of cold water into a steam-filled pipe are less than those calculated for the design basis pipe break. They have cited two tests (17, 18) which are 1/14 and 1/3 scale tests, respectively. In these tests, which examined the interaction of steam with cold injection water, the maximum observed pressure fluctuations were 40 psi for the 1/14 scale test, and 18 psi for the 1/3 scale test.

Two other sources of experimental data, the LOFT and Semiscale facilities, were examined by EG&G at the Idaho National Engineering Laboratory in Idaho. The result of the review (19) of the data from these facilities was that fluid oscillations could occur, but that the pressure oscillations are "small (less than 10 psi) to non-existent.

Of significant concern during the injection of cold water into steam in the potential for "slugging," in which the liquid flow entering the pipe is sufficient to fill the cross-sectional area of the pipe. Condensing steam on the surfaces of this water slug sets up the pressure gradients necessary to initiate oscillations of the slug in the pipe.

This oscillating liquid slug not only gives rise to pressure oscillations as observed in the tests, but it can also produce inertial loads at pipe bends, and loads if it impacts on surfaces (e.g., a core barrel).

The tests cited by W in References 17 and 18 were not representative of an entire PWR system. Moreover, they were preformed at low system pressures more representative of accumulator injection.

Calculational models designed to predict these pressure oscillations due to water slugs indicate that the amplitude of pressure oscillations should not increase as the pipe diameter increases, as long as all other factors (such as system pressure, length-to-diameter ratio, etc.), do not change. Moreover, the amplitude of these pressure oscillations is predicted to increase as system pressure increases.

The previous discussion applies to cases when the pipe is filled with a water slug. If the cold water injection flow is insufficient to produce a water slug that fills the cross-sectional area of the pipe, then the above-described phenomena are not expected to occur.

Based on the above, we cannot conclude that the tests cited by W support their conclusion that the pressure oscillations are accommodated in the structural design.

We require that W provide information to confirm that HPI and accumulator flows during small breaks will not result in the formation of water slugs, or if they do, to show that the structural design bases of the primary system include loads due to:

(a) water slug inertial motion
(b) water slug impact
(c) pressure oscillations due to steam condensation.

Any test data cited must be shown to be applicable to the actual system design. We intend to pursue this issue, in detail, as part of the generic safety review of water hammer in nuclear power plants, Task Action Plan A-1 (40).

4.1.5 Break Enthalpy is Not Core Exit Enthalpy (Generic, from Reference 7)

The concern as expressed by TVA dealt with the possible bypassing of the core by injected HPI water, and the discharge of HPI water out of the break. If the break flow energy was not representative of the core exit energy, then less than the maximum percentage of the decay heat load would be removed by the break.

The W small break evaluation model considers fluid enthalpy distribution and transport within the system. In considering the possibility of injected HPI water bypassing the core and discharging out the break, the W model assumes that none of the HPI fluid expected to be injected into the broken leg (for cold leg breaks) is injected. For cold leg breaks, the amount of HPI water injected into the cold legs of the intact loops that travels around the downcomer annulus and exits the break in the broken loop without entering the core, is also accounted for, using a drift-flux model.

Based on this concern, as well as on previous vendor system pressure predictions for Semiscale Test S-02-6, the staff is performing an ongoing evaluation of small break system behavior during ECC injection.

The scope of this evaluation includes both injection location modeling and system pressure behavior during injection.

It is expected that the adequacy of the current models will be determined by comparison to Semiscale Test S-07-10B and the forthcoming LOFT Small Break Test L3-1 (see Section 4.2.1.5 of this appendix).

We conclude that, while the W evalution model accounts for the fluid enthalpy distribution and transport in the primary system, as well as injected HPI fluid bypassing the core, the overall adequacy of the predicted system behavior during injection of emergency core coolant should be confirmed by comparison to experimental data.

4.1.6 Long-Term Source of Auxiliary Feedwater

The assurance of "a continuous long-term source of clean auxiliary feedwater for the steam generators...," was raised by TVA as a potential concern if the recovery should be delayed pending additional fuel cooldown.

In W-designed plants, the immediate source of auxiliary feedwater is the condensate storage tank. Westinghouse has estimated that this tank can supply auxiliary feedwater to the steam generators for seven to ten hours, depending upon the particular plant design. For most small break LOCA scenarios, this supply is expected to last well beyond the recovery period of the accident, and should provide ample time for the normal feedwater to be restored.

In the event of degraded conditions during a small break LOCA, in which the recovery period is extended beyond the above-stated time, the condensate storage tank can continue to supply auxiliary feedwater (usually service water).

In the emergency guidelines for small break LOCAs for W-designed plants, the W Operating Plants Owners Group noted that the condensate storage tank should be monitored by the operators, and that switchover to an alternate source of water should be performed when the tank indicates a low level.

Based on the design of W plants, which provides an alternate unlimited backup source of auxiliary feedwater, and the guidelines for emergency procedures alerting the operator to monitor the auxiliary feedwater supply, we conclude that adequate supplies of auxiliary feedwater are available and that the TVA concern has been adequately considered.

4.1.7 Recirculation Mode of High Pressure Safety Injection Operation at High Pressures not an Established Desigh Requirement

This concern was based on the fact that, for some CE-designed plants, the HPI pumps have a shutoff head of about 1300 psig. If decay heat removal could not be accomplished due to disruption of natural circulation, the system would repressurize. If the system then remained at high pressure for an extended period, the recirculation mode for HPI operation would eventually be activated. TVA's concern was the availability of a HPI system designed to operate in the recirculation mode with the reactor at high pressure, and the evaluation of such operation for feasibility.

TVA also had a related concern regarding minimum flow protection for the HPI pumps. This is addressed in Section 4.1.14 below.

With regard to operation of the pumps at high pressure, these pumps cannot supply emergency core cooling water to the primary system if the system pressure rises above their shutoff head. System pressure can only rise above the shutoff head if there is degraded or no secondary heat removal. In this event, the plant operators are relied upon to open the PORVs in order to depressurize the system below the shutoff head of these pumps and enter into a "feed and bleed" mode of operation.

With regard to extended operation at elevated pressures below the shutoff head and minimum flow requirements, W has confirmed that this mode of operation is a design requirement for their plants as specified in the Nuclear Steam Supply System (NSSS) Standard Information Package (SIP) supplied for each plant.

We have concluded that, in the event of degraded secondary heat removal or failure to establish natural circulation, operator action is necessary to open the PORVs and depressurize the system below the shutoff head of the HPI pumps. In addition, design requirements exist for extended pump operation at pressures below that required to meet the minimum flow requirements.

4.1.8 Use of HPI Pumps with the Shutdown Cooling System

For both CE- and W-designed plants, the long-term cooling after recovery from a small break LOCA is performed by a low head, high volume cooling system (termed shutdown cooling system (SCS) for CE-designed plants and Residual Heat Removal (RHR) system for W-designed plants). This system circulates primary coolant to heat exchangers, cools it, then returns it to the primary system. For CE-designed plants, inventory in the primary system during this period must still be maintained by the HPI pumps. TVA expressed a concern about the simultaneous use of these two systems (HPI & SCS), "since it involves some common piping including a common return pipe." Moreover, TVA was concerned that "this mode of operation has also not been a design requirement."

For all W-designed plants, there is a basic design difference from the CE System 80 design, which does not make this concern directly applicable. For CE System 80 plants, the SCS and the HPI pumps share a common suction from the refueling water storage tank during the injection mode. During the post-accident cooling modes (initial recirculation, simultaneous injection, and shutdown cooling), the HPI and SCS pumps switch suction from the refueling water storage tank to other sources. The HPI switches suction to the containment sump, while the SCS switches suction to the hot legs. Any cross-connections or common piping shared during the injection mode must be removed from the HPI suction path after the injection mode to prevent loss of Net Positive Suction Head (NPSH) due to common suction.

In W-designed plants, the HPI pumps take suction from the RHR pumps in the recirculation mode for long-term heat removal. Westinghouse states that "Runout and NPSH protection is assured by design criteria which specify adequate NPSH, assuming minimum resistances in the RHR outlet piping and maximum (runout) flow on the high heads."

We conclude that the TVA concern is not applicable to W-designed plants, because of basic differences between the RHR-HPI suction piping arrangements and the SCS-HPI piping arrangement for the CE System 80 design.

4.1.9 Mechanical Effects of Slug Flow on Tubes

Once the high pressure safety injection system begins to add mass to the primary system at a rate greater than that of the break flow out of the system, the system inventory will stop decreasing and begin to increase. This is what TVA termed the recovery period in Reference 6. If the liquid level dropped below the hot leg outlets, decay heat removal via pool boiling/steam condensation would have been established As the vessel liquid level increases during the recovery period, the hot legs would become blocked with liquid, disrupting steam flow from the core to the steam generator. Vapor generated in the core would accumulate in the vessel upper plenum and upper head with a subsequent rise in pressure. This pressure rise would force the vessel liquid level down until a venting path for the steam from the vessel to the steam generator was established. TVA's concern was that this condition would produce slug flow in the hot leg pipe, resulting in mechanical loadings on the steam generator tubes.

Westinghouse, in response to this concern, has examined the dynamic loadings that result from the subcooled blowdown portion of the design basis LOCA (double-ended guillotine break of the cold leg at the steam generator outlet nozzle) for which the steam generators are qualified to maintain their structural integrity. These dynamic loadings were reported to be primarily due to the rarefaction wave which produces a pressure from 400 to 800 psi, depending upon the plant. The 800 psi pressure was calculated to produce a stress in the steam generator tubes of 48,000 psi, assuming 40 years of tube service.

While not reporting any specific values, W has concluded that any loads due to slugging would be less limiting than the design basis LOCA loads.

We have performed a preliminary analysis on the possibility of steam slugging during the hot leg recovery period. Flow regime maps in available literature indicate a stratified flow regime would exist for the expected steam flow conditions. However, these flow regime maps were developed from long pipe flow data and have not been shown applicable to short pipes in which entrance effects could be significant. As such, the possibility of slugging cannot be eliminated at this time.

Westinghouse has stated that they will continue to examine this phenomenon. We will require that Westinghouse submit the results of this examination to confirm their conclusion that slugging loads would be bounded by the design basis LOCA loads. Should this examination result in calculated loads due to this phenomenon exceeding the design basis loads for the steam generator, appropriate action will be taken to assure structural integrity of the steam generators under all design basis conditions.

4.1.10 Effect of Safety Injection Tank Discharge/Non-Condensible Gas Accumulation in the System

The concern expressed by TVA is that the effect of the safety injection tank discharging into the system, once the pressure drops below approximately 600 psig, has not been evaluated. Since these tanks are pressurized with nitrogen gas, there is the potential for injecting this gas into the system. Introduction of non-condensible gas into the primary system has the potential to disrupt natural circulation and reduce steam generator condensation heat transfer.

Westinghouse has responded to this concern by stating that, for breaks smaller than four inches in diameter in three-loop plants, the accumulators either will not inject, or will not inject completely. Therefore, none of the nitrogen used to pressurize the accumulators is calculated to enter the system. Westinghouse analyses also show that, for breaks greater than two inches in diameter, the break alone is calculated to remove the decay heat such that the steam generators are not required for decay heat removal. Thus, for any breaks (greater than four inches in diameter) in which accumulator nitrogen can enter the primary system, disruption of natural circulation and reduction in steam generator heat transfer are not of concern, since the steam generators are not required for decay heat removal.

For those small breaks in which the accumulator is calculated to inject (greater than two inches for a three-loop plant), the WFLASH evaluation model computer code will calculate the effects of this injection.

The staff has expressed a concern about the ability of the analytical models to correctly predict the ECC injection behavior, in particular, the effects of injection on system pressure and subsequent effect on the amount of injected ECC water. This concern is discussed in more detail in Section 4.2.4 of this appendix.

For those small breaks which do not depressurize to the accumulator setpoint, recovery and long-term cooldown procedures should instruct the operator to isolate the accumulators before depressurizing below the accumulator setpoint pressure to preclude accidental injection.

In Section 4.2.1.4 of this appendix, a more complete discussion is provided on sources and effects of non-condensible gases in the primary system.

We conclude that the initiation of accumulator injection is properly accounted for by the W evaluation model. During the longer-term recovery not explicitly calculated with the evaluation model, approved emergency operator guidelines require that the operators isolate the accumulators before depressurizing below their injection setpoint.

4.1.11 Pump and/or Pump Seal Damage/Continuous Operation of Pumps

The concerns expressed by TVA regarding pump and/or pump seal damage and continued operation are that continued operation during depressurization could cause pump cavitation, and associated vibration could lead to mechanical damage of the pump itself or the pump seals.

Moreover, if offsite power is assumed to be lost, TVA was concerned that failure to either restore offsite power, or to load the chemical volume and control system on the diesel generators could also result in pump seal failure.

4.1.11.1 Pump Seal Damage

Westinghouse has stated that seal injection flow will be maintained to the pumps during an accident, even if offsite power is lost. In the event offsite power is lost, the pumps are designed to be tripped and seal damage could not occur. In addition, the recently approved W generic guidelines for emergency procedures require the operator to trip the reactor coolant pumps after high head safety injection pump operation has been verified, and the reactor coolant pressure is below a specified setpoint*. Moreover, these guidelines instruct the operator to trip all reactor coolant pumps within five minutes, if component cooling water to the pumps is lost.

Westinghouse has not provided an analysis assuming a small primary system break in conjunction with pump seal failure. They have stated that in the event of seal leakage during a small break, it would appear as a slightly larger break and would therefore be bounded by present FSAR analyses.

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*The detailed method of how this setpoint is obtained is documented in Section 7.2.2 of Reference 4. It is expected that the setpoint will be about 1400-1600 psig.

Because no confirmatory analyses have been presented, we cannot conclude that the accident would appear as a larger break and be bounded by FSAR analyses. We will require W to provide the results of an analysis assuming the worst credible failure of the pump seals in conjunction with a small break occurring at the worst location to confirm their conclusions, or to demonstrate why seal failure concurrent with a small break LOCA should not be considered as an accident scenario.

4.1.11.2 Continuous Operation of Pumps

Recent preliminary analyses by each of the three PWR vendors to determine the effects of (1) continuous reactor coolant pump operation throughout the entire small break LOCA, and (2) tripping the pumps part way through the accident, have produced results that indicate failure of the pumps part way through a small break accident under certain conditions could lead to cladding temperatures which exceed acceptable limits. In addition, one vendor has performed analyses of a hot leg break which result in unacceptable consequences, even if the pumps remain running throughout the entire accident.

As a result, the staff issued Office of Inspection and Enforcement (IE) Bulletins 79-05C and 79-06C on July 26, 1979, which instruct operators to shut off the reactor coolant pumps upon reactor trip and initiation of HPI caused by low reactor coolant system pressure. Reports by all the PWR vendors on the effect of delayed tripping of the pumps during a small break LOCA were submitted by the owners groups in response to these bulletins. Our review of these reports and the conclusions resulting therefrom are documented in Reference 4.

4.1.12 Initiation of Containment Spray

In the event of a very small break LOCA, TVA was concerned that a low pressurizer level would occur early in the accident and initiate containment isolation. Subsequent high containment pressure would then initiate containment spray. In particular, a prolonged isolation due to delay in cooldown would increase the likelihood of initiating containment spray.

Westinghouse has stated that all of the equipment inside of containment necessary to mitigate the accident and safely shut down the plant has been qualified for the post-small break LOCA environment, including the effects of containment spray.

Recent W analyses, however, have shown that the reactor coolant pumps must be off within 10 minutes of the start of a small break LOCA, or the tripping of the pumps later in the accident could result in peak cladding temperatures in excess of the 2200°F limit specified in §50.46. Also, pump restart criteria and requirements have not yet been identified. The ability to restart the pumps is dependent, in part, upon the post-LOCA environment. This will be studied during our review of inadequate core cooling, as defined in NUREG-0578.

Moreover, we have identified other equipment located inside of containment, currently not designed to perform in the post-LOCA environment; but nevertheless relied upon and its proper operation assumed during the analysis of small break LOCAs. This equipment includes items such as the pressurizer spray valve, and the pressurizer PORVs. We expect to obtain definitive resolution of this concern in the implementation of Recommendation 9 of NUREG-0585.(5)

4.1.13 Reactor Vessel Coolant Level Unknown to Operators/Adequacy of Emergency Operating Procedures

During the course of a small break LOCA, the mode of decay heat removal will switch from single-phase liquid natural circulation to pool boiling/condensation and then back to subcooled liquid natural circulation. As was discussed in Section 4.1.9 in this appendix, the system hydraulic behavior could become erratic during the transition due to slug flow, and the operators could observe fluctuations in system pressure and vibration, to name a few possible symptoms. TVA's concern was that, during this period of potentially unstable system hydraulic behavior, the operators would not have vessel level indication, and without clear emergency procedures to guide them, could take the wrong or inappropriate actions.

Westinghouse has responded to this concern by pointing out that no operator action is required during this phase of a small break LOCA, except to verify that the reactor is tripped, safety injection and auxiliary feedwater are initiated, and the reactor coolant pumps are tripped.

We agree that system behavior could become erratic during this period. This does not pose a problem, however, provided that the potential erratic behavior does not precipitate incorrect operator actions. In particular, the operator should not terminate or degrade safety injection, or otherwise jeopardize the normal sequence of actions of plant safety features. It is our conclusion that these and other necessary precautions have been properly incorporated in plant emergency procedures, and, therefore, no operator action will be needed during this phase of a small break LOCA. Therefore, incorrect or inappropriate operator action will not be precipitated.

4.1.14 Minimum Flow Protection for HPI Pumps During the Recirculating Mode of Operation

The high pressure injection pumps have a minimum flow requirement (about 30 gpm) to insure adequate pump cooling and preclude damage. During the injection phase of a small break LOCA, the HPI pumps take suction from the refueling water storage tank. Minimum flow is assured by a "miniflow" line from the pump discharge back to the refueling water storage tank. Once the refueling water storage tank is sufficiently depleted, the HPI pump is switched over to take suction from the containment sump. This is called the recirculation mode.

When in this mode of operation, there is no minimum flow protection for the HPI pumps in the event the system pressure rises above that pressure which allows minimum flow in the pump.

Although for expected small break LOCA conditions, pump flow is calculated to be sufficiently above the minimum flow value, under certain conditions (e.g., an isolated break), the system pressure could rise back up above the pressure needed for minimum flow protection.

Current W emergency procedure guidelines instruct the operator to observe whether the system pressure is above the shutoff pressure of the pumps, and if it is, to shut off the pumps.

The W generic guidelines caution the operator to be alert for the possibility of pumping against the safety injection pumps' shutoff head during the recirculation mode of operation. Since operator involvement in long-term core cooling will vary, depending on plant ECCS design and available instrumentation, justification of the operator's capability to respond properly to this condition will be pursued on a plant-specific basis.

4.1.15 Conclusions

Our conclusions regarding the TVA concerns about small break LOCAs discussed in the preceding sections are as follows:

1. Pressurizer level is a valid indication of system inventory only when it can be demonstrated that no voids exist in the primary coolant system.

2. Safe recovery from small break LOCAs can most reliably be achieved by isolating the break, if possible.

3. If feedwater is not available, whether or not a break is isolated, operator action is necessary to manually open the PORVs and depressurize the system such that adequate HPI can be initiated.

4. Adequate supply of auxiliary feedwater exists in the event of a delayed recovery from a small break. (See Appendix X.)

5. We cannot conclude that slug flow will be avoided in the hot leg during small break recovery. Additional analyses by W addressing this matter should be completed and the results submitted for staff review.

6. We have concluded that the W evaluation of non-condensible gas effects is acceptable, but some confirmatory information will be needed.

7. The small break LOCA models have not been adequately compared against integral systems tests. Such comparisons are in progress.

4.1.16 Recommendations

Recommendations based on our review of W's consideration of the TVA concerns and the conclusions reached above are as follows:

1. Emergency procedures are required to instruct operators of the need to open the PORVs in the event that feedwater is not available when a break is isolated.

2. An analysis of a break in the pressurizer spray line in which the isolation valve is assumed to fail open should be performed.

3. Emergency procedures for small break LOCAs should instruct the operator that loop isoltion valves should not be used to isolate the breaks.

4. Information is required to confirm that HPI and accumulator flows during small breaks are insufficient to form water slugs, or if they do, to show that the structural design bases of the primary system include loads due to:

   (a) water slug inertial motion,
   (b) water slug impact, and
   (c) pressure oscillations due to steam condensation.

5. Small break emergency procedures should instruct the operators of the need to isolate the accumulator tanks during recovery from a small break.

6. Analyses are required assuming the worst failure of the pump seals in conjunction with a small break in the most limiting locaton, or additional justification is necessary to demonstrate that pump seal failure is precluded.

7. Additional analyses must be provided on the effect of spray isolation valve failure on small break LOCAs. Non-equilibrium effects must be considered if they are important to the results.

8. Additional analyses must be provided to confirm that, for conditions in which counter-current flow is predicted to be achievable in the pressurizer surge line, a steam source from the hot leg would not be available.

9. Verification is required for the following: (1) predicted condensation heat transfer rates in the presence of non-condensible gases, and (2) the W conclusion that the effect of non-condensible gas on system pressure and consequent system behavior is negligible.

10. We require that W provide formal documentation of the core uncovering test used in support of the vessel mixture level model for small break LOCA analyses.

4.2 Small Break LOCA Analysis

4.2.1 Analytical Model

4.2.1.1 Analytical Methods

Westinghouse performed a series of small break calculations using the NRC-approved small break evaluation model. This model is described in WCAP-8200, Revision 1 (July 1974), and WCAP-8970 (April 1977), and is normally used to perform small break LOCA analyses for postulated break areas up to 0.5 square foot. The calculations discussed in the appendix have focused on the very small break spectrum and include consideration of breaks in the pressurizer vapor space. These calculations are beyond the scope normally considered in small break LOCA analyses, and, for this reason, we expressed a number of concerns about the applicability of the current models. Our concerns were as follows:

(1) Following postulated small break LOCAs, a primary mechanism for heat removal is natural circulation. We were concerned about the ability of the computer programs to correctly predict the various modes of natural circulation and the interruption of natural circulation, if it should occur. We noted that experimental data for the verification of methods for two-phase natural circulation are currently not available.

(2) The experimental verification of small break LOCA analysis methods with systems data was limited. The available small break data from the Semiscale facility (Test S-02-6), although containing a number of deficiencies, represented the best information then available. The analytical methods used to predict the results of this test did not correctly predict the overall system depressurization rate, and the depressurization rate following injection from the safety tank. These were significant parameters in that they affected the injection rate of the safety injection fluid. Improved data from the Semiscale (Test S-07-10B) and the LOFT (Test L3-1) facilities should be available for model verification by the end of 1979.

(3) The appropriateness of the pressurizer model for analyses of small breaks at various locations was a potential concern. It was noted that the equilibrium pressurizer model assumed in the vendor analyses gave somewhat different results from hand calculations assuming non-equilibrium conditions. These modeling differences may be significant for various postulated breaks. Also, the representation of potential flooding in the surge line could affect the draining of the pressurizer. A flooding check is not made for the surge line in the

computer programs. Thus, potential pressurizer draining was a concern in these analyses.

(4) The calculation of core level and core heat transfer are important features of the small break LOCA model. Limited experimental data was then available to verify these models. Although the comparisons available at that time had been satisfactory, the data were not challenging to the codes. It was anticipated that more data would be obtained for further code verification.

(5) The number of nodes used to represent the primary system for small break LOCA analyses should be sufficiently detailed to model the flashing of hot fluid in various locations. This modeling detail is necessary since the calculated system pressure during the decompression process is controlled by the flashing of the hottest fluid existing at any time in the model. The assumption of thermal equilibrium requires that the fluid combined in a single node be represented by the average fluid properties. If fluid from several regions is combined in one node, the calculated system pressure during a portion of the transient may be lower than could occur if the smaller regions of hot fluid flashed and maintained the system at the corresponding saturation pressure. Thus, the modeling detail could have a significant effect on the calculated parameters of interest for various events such as ECCS actuation.

(6) During the period of recovery from a small break LOCA, the thermodynamic equilibrium assumed in fluid control volumes could result in errors in the predicted system pressure. This could, in turn, introduce errors in both the break discharge flow and the safety injection flow. The specific concern involves the rate at which the water, which is refilling the system can condense steam. If the condensation efficiency is less than 100 percent, then system pressure would be higher than that predicted.

(7) The discharge rate of two-phase fluid through the relief and safety valves is an important consideration for some transients. These include postulated stuck-open relief or safety valves, and primary system depressurization for very small or "zero break" LOCAs by opening the relief valves if a "feedwater is assumed lost. There is a lack of discharge rate data for two-phase fluid at high pressure; most available experimental data is for steam at low pressure.(21) If the actual valve discharge flow is lower than that assumed in the calculations, primary system depressurization to the high pressure injection set point might not occur within the calculated times.

To address the above concerns regarding the small break LOCA model, W performed under; the sponsorship of the W Operating Plants Owners Group, additional noding studies and evaluations to justify various aspects of the model. This additional information was discussed with the staff at a meeting on June 11, 1979, and subsequently documented in WCAP-9600, Section 2.0.(3) These topics are discussed in the following sections.

4.2.1.2 Pressurizer Model

In the TMI-2 accident, the pressurizer behavior played a key role in influencing the operators' actions. In particular, the insurge of fluid to the pressurizer led to incorrect operator diagnosis of system inventory. Because of this, we concluded that it was appropriate to reexamine the analytical modeling of the pressurizer to determine if predicted behavior for other similar accidents would represent realistic behavior, and if it could be used by the operators for accident diagnostics.

This reexamination led to staff concerns which we expressed about the following:

(1) the effect of nodalization.
(2) the effect of initial hot leg subcooling.
(3) the effect of both pressurizer sprays and heaters.
(4) the pressurizer surge line representation.
(5) the pressurizer draining behavior.
(6) the effect of equilibrium model assumption during periods of flow insurge to the pressurizer.

In response to our concerns, the W Owners Group provided analyses regarding items (1) through (4). Discussions between the staff and the W Owners Group were held regarding items (5) and (6).

A summary of our review and conclusions regarding these analyses follows.

4.2.1.2.1 Effect of Nodalization

In order to determine the effect of nodalization, W examined the effect of increased pressurizer nodalization from one to four nodes on the analysis of two small break LOCA's. These two accidents were pressurizer vapor space breaks of 0.008 ft^2 and 0.034 ft^2, which correspond to the flow areas of one and three power-operated relief valves (PORVs), respectively. It should be noted that the three PORVs are not of equal size, which is why the flow area of three valves is not equal to three times the flow area of one valve.

The one-node case utilized a bubble rise model, whereas the four-node case utilized a homogeneous model for the lower three nodes, and a bubble rise model for the top node (which included the vapor space).

For both of the accidents analyzed, differences between the one- and four-node models were shown to be very small, having a negligible impact on the overall system behavior. This was illustrated by comparisons of differences in the integral break flow and break energy at 1000 seconds into the analysis. For the 0.008 ft^2 break, the four-node case showed an increase of 0.2% in integral break flow, and 0.07% in integral break energy at 1000 seconds over the one-node case.

Similary, for the 0.034 ft^2 break, the four-node case showed an increase of 0.22% in integral break flow and 0.1% in integral break energy at 1000 seconds over the one-node case.

4.2.1.2.2 Effect of Initial Hot Leg Subcooling

In the base case examined by W, the hot leg was approximately 45 Fahrenheit degrees subcooled, compared to the pressurizer. To determine if system behavior differences due to pressurizer nodalization were affected by the magnitude of the hot leg subcooling, W performed an additional analysis in which the hot leg was initially 60 Fahrenheit degrees subcooled, compared to the pressurizer. For the 60° subcooling case, the integral break flow for the four-node pressurizer model was 0.056% less than the one-node model, and the integral break energy was 0.05% less than the one-node case. For this study, the 0.034 ft^2 break was used, since it would be expected to exaggerate any differences.

4.2.1.2.3 Effect of Pressurizer Sprays

The presurizer spray line is a line from the cold leg piping to the top of the pressurizer. It is used to reduce system pressure during normal control operation by spraying the cooler cold leg water into the pressurizer vapor space and condensing steam. When pressure is decreasing, the spray lines are automatically isolated, and the pressurizer heaters are turned on. For increasing pressure, the sprays are automatically turned on. During a small break LOCA, introduction of a subcooled spray into the pressurizer steam space could result in significant non-equilibrium conditions, invalidating present analytical model equilibrium assumptions.

Westinghouse has stated that non-equilibrium effects due to pressurizer sprays need not be considered, since during any LOCA, the pressurizer pressure will decrease initially, thus, automatically isolating the pressurizer spray line. In addition, emergency procedures instruct the operator to verify that all pressurizer spray lines are closed and to close them, if they are open.

We agree that, under expected conditions, the pressurizer spray line would be isolated and, therefore, would not affect currently predicted behavior. However, failure of the spray isolation valves to close could affect a certain class of small break LOCAs, such as pressurizer steam space breaks. We, therefore, require that W provide additional analyses showing the effect of spray isolation valve failure on small break LOCAs. If non-equilibrium effects are considered to be important to the results, they should be appropriately considered in the analyses.

4.2.1.2.4 Effect of Pressurizer Heaters

During normal operation, whenever the system pressure is below the control system set point, the pressurizer heaters are automatically turned on to raise the system pressure.

However, the heaters are automatically shut off if the level in the pressurizer drops below a minimum level. For all breaks located below the pressurizer elevation, the pressurizer is calculated to drain and the heaters are expected to turn off.

In the event that pressurizer draining does not occur as predicted, the pressurizer heaters could remain on. This would have the effect of maintaining the pressure in the pressurizer and thus force liquid in the pressurizer to drain due to a maintained higher pressure.

Since pressurizer draining is beneficial in the sense that it transfers liquid from the pressurizer to the primary system (particularly the core), failure of the heaters to shut off would enhance draining and thus the primary system inventory. Therefore, for the case of breaks located below the pressurizer, assuming that the heaters turn off is conservative.

If the break was in the surge line, maintaining the heaters on would also enhance draining and provide the operator with a quicker indication of inventory loss. Therefore, we agree that the impact of heaters on small breaks located below the pressurizer elevation is negligible. The case of a break due to a stuck-open PORV was considered to be the more sensitive case, since for this event, the pressure is dropping, yet the pressurizer maintains a significant liquid inventory. Thus, the control system would maintain power to the heaters during the event.

The effect on system thermal-hydraulic behavior due to representing the pressurizer heaters in the system evaluation model was examined for both the one- and four-node pressurizer models.

With respect to the previous-discussed sensitivity studies discussed, the pressurizer heater study showed the largest sensitivity of system behavior to pressurizer nodalization. However, the differences were still not significant and show that the current one-node pressurizer representation is conservative compared to the four-node representation. This is because the one-node representation predicted a higher integral break flow, and thus a greater system mass depletion than the four-node representation.

For the 0.008 ft^2 case, at 1000 seconds, the integral break flow was 10.46% less for the four-node representation than the one-node representation. Similarly, the integral break energy was 3.733% less for the four-node representation at 200 seconds (at 500 seconds, both integral break flow and integral break energy were about 2% higher). For the 0.034 ft^2 break, the integral break flow and integral break energy were 1.38% and 0.516% less for the four-node representation than the one-node representation at 1000 seconds.

4.2.1.2.5 Effect of Non-Equilibrium During Flow Insurge to Pressurizer

The condition which would result in the most highly non-equilibrium conditions existing in the pressurizer is considered to occur when an insurge of liquid enters a steam-filled pressurizer, which does not have a vent path to relieve steam. For this condition, the entering liquid would not condense steam with 100% efficiency. This is due to a number of considerations, primarily:

1. The establishment of a saturated or near-saturated liquid layer at the steam-liquid interface, which would retard condensation.
2. The rate of condensation heat transfer at the surface may not condense steam at a rate high enough to prevent pressurization.
3. Non-condensible gases in the pressurizer would reduce condensation heat transfer rates.

The result of these considerations is that the equilibrium models would overpredict the refilling rate of a pressurizer and underpredict the pressure. While W̲ has not provided an analysis of this effect, we do not consider this to be a problem. The only times such an insurge have been identified are during the recovery period of a small break LOCA and for pressurizer breaks in which liquid insurge into the pressurizer will be experienced. However, for pressurizer breaks, the system is depressurizing and the mode of energy transfer will be liberation of latent heat (flashing) rather than condensation of steam. As such, non-equilibrium concerns are not considered applicable.

During the recovery period, non-equilibrium behavior would tend to make the actual pressurizer filling slower than an equilibrium model prediction. This would not precipitate incorrect operator action such as prematurely terminating HPI flow, however, since the approved HPI termination criteria are keyed on rising level and not on that entirely.

While we conclude that non-equilibrium processes in the pressurizer during refill will not affect the action taken by the operators, we will require that the non-equilibrium behavior be examined in greater detail as part of the evaluation model review for compliance with Appendix K to 10 CFR Part 50.

4.2.1.2.6 Pressurizer Surge Line Representation

The W̲ small break LOCA evaluation model represents the surge line connecting the hot leg to the pressurizer as a simple pipe. The flow in the pipe is determined from the momentum equation, and a critical flow check is performed by the evaluation model code. Under certain conditions, it can be postulated that counter-current flow would be established in the surge line, which could allow the pressurizer liquid to drain while hot leg steam flowed into the pressurizer. To determine if this condition was occurring, W utilized a drift flux model in the pressurizer surge line representation to better represent the flow characteristics. In particular, this model would predict the onset of counter-current flow, should it exist.

A calculation using Appendix K to 10 CFR Part 50 assumptions with a pressurizer vapor space break of 2.5 inches in diameter was performed. This calculation showed that a flooding* condition existed in the surge line throughout the entire transient, and, therefore, no counter-current flow was predicted.

Westinghouse extended their analyses to breaks of other sizes by hand-calculated solutions of the drift-flux relationships for a range of pressures from 1000 to 2000 psia.

Based on these calculations, W concluded that a narrow range of pressurizer break sizes exists (with the lower bound being that break size for which makeup flow would prevent system depressurization) for which counter-current flow is calculated to occur, and therefore, pressurizer draining is predicted.

Since evaluation model two-phase critical flow rates are predicted to be greater than best-estimate flow rates, use of best-estimate methods would have the effect of raising the minimum break size in which counter-current flow would be predicted.

The hand calculations performed by W assumed that steam flow was available from the hot leg. For very small pressurizer steam space breaks, W believes that the system will be essentially water-solid, with very little steam available from the hot leg to allow counter-current flow. We will require W to submit additional analyses to confirm that this is the case.

A pressurizer break (such as a stuck-open relief valve) could possibly occur in which counter-current flow was established in the surge line. This would appear to the operator as a small break in the primary system. Thus, the operator might not initially diagnose the problem as a pressurizer break and more specifically as a possible stuck-open valve in which isolation would be desired.

The consequences of stuck-open valves have been evaluated and shown to be acceptable. The possibility of counter-current flow and pressurizer draining could possibly result in misdiagnoisis of break location and delay break isolation of those breaks which could be isolated (such as a stuck-open valve). Other symptoms, however, such as relief valve position indication, tailpipe temperature, and quench tank pressure should provide the operator with enough diagnostic information to take appropriate corrective action.

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*Surge line steam flow into the pressurizer was of sufficient velocity to keep water from draining out of the pressurizer via the surge line.

4.2.1.2.7 Conclusions

Westinghouse has presented an evaluation of system behavior sensitivity to certain pressurizer modeling assumptions. We have reviewed these analyses and have concluded that, while the analyses presented support the appropriateness of the present evaluation model pressurizer model for predicting system response to small breaks in the primary coolant system, additional confirmatory information is needed in certain analytical areas.

4.2.1.3 Steam Generator Model

Each U-tube steam generator model contained in the approved small break evaluation model consists of three nodal volumes: a single secondary side volume, and two primary side volumes, which represent the hot and cold sides of the U-tubes. The secondary side and U-tube cold side volumes allow phase separation by steam bubble rise when two-phase conditions exist, while the U-tube hot side volume assumes a homogeneous state at all times.

Two sensitivity studies were conducted, one to assess the effects of additional nodal detail in the tubes, and the other to assess the assumption of homogeneity in the hot side of the U-tubes. Doubling of the hot and cold side U-tube noding, while keeping the same fluid state assumptions on both sides, had very little effect on the outcome of the core uncovering transient for a two-inch diameter cold leg break in a four-loop plant. As shown in Table VIII-5 for Cases A and B, the intervals of core uncovering are identical, but with a slightly deeper uncovering transient for the two-node (approved) model, which would show a slightly higher peak cladding temperature (PCT) if cladding temperatures were computed.

Allowing phase separation in the U-tube hot side of the two-node model results in water drainage to the reactor vessel, thereby delaying and reducing core uncovering, as shown for the results given for Case C. The same U-tube hot side assumption for the four-node model in Case D did not, however, show as much hot side drainage and increase in time to the start of uncovering of the core, but did show less core uncovering than Case B. Use of phase separation in the hot side of the four-node steam generator U-tube representation led to vertical stacking of mixture and steam spaces on the hot side, so that drainage was delayed, producing the earlier core uncovering, but resulting in the reduced depth of uncovering.

From these results, we agree with W that for cold leg breaks, the two-node U-tube representation of the approved model coupled with the required homogeneous assumption on the tube hot side provides conservative core uncovering predictions for small breaks resulting in depressurization to the secondary side pressure level or below. However, this nodal representation has not been compared with alternate nodal models for the effect on natural circulation predictions, and is possibly non-conservative for hot leg breaks. The homogeneous assumptions used on the U-tube hot side will

TABLE VIII-5

STEAM GENERATOR MODELING STUDY

Case	No. of U-Tube Nodes	Hot Side Fluid Model	Core Uncovering		
			Start(sec)	Depth(ft)	Interval(sec)
A	2	Homogeneous	1750	5.5	140
B	4	Homogeneous	1750	6.3	140
C	2	Heterogeneous	1950	7.5	130
D	4	Hetrogeneous	1640	8.4	80

provide a lower quality fluid in the steam generator U-bends compared to a phase separation model for this node. As a result, an alternative model which considers phase separation in the U-bend region as well as U-bend elevation effects could be expected to result in differing predictions regarding interruptions of natural circulation during a depressurization transient. These differences would influence the conditions of the fluid near cold or hot leg breaks, thereby influencing system coolant inventory and core covering in small break LOCAs.

The two-phase mixture behavior in the tubes under condensation heat transfer conditions is believed to be more complex than either model can describe. Experimental verification of the phenomena will be required to justify the use of one or the other model, or a new model.

Overall steam generator heat transfer between the primary system and the secondary system is modeled as the sum of four heat transfer modes over the U-tube length. These modes consist of steam-to-steam, steam-to-liquid, and are proportioned according to primary and secondary mixture heights. Primary side liquid region heat transfer is determined as a function of flow rate, while vapor region heat transfer is determined as a function of the primary system to secondary system temperature gradient modified by degradation effects introduced by the presence of non-condensible gases (see Section 4.2.1.4, which follows). Secondary side liquid and vapor region heat transfer are represented by fixed values, based on pool boiling in the liquid region and natural convection in the vapor region.

Computation of the overall heat transfer rates between the primary and secondary systems during LOCA transients using these heat transfer modes is considered adequate. However, further experimental verification of the application of these heat transfer modes to this condition is required. The conservative assumption imposed on reactor vessel inventory by the requirement to model the U-tube hot side as a homogeneous mixture effectively reduces the sensitivity of system response to steam generator heat transfer modeling, as has been demonstrated by W in earlier perturbation studies. As a result, the existing heat transfer modes contained in the approved model are not fully utilized for the small break studies, since the assumption of a homogeneous U-tube hot side model does not permit the proper application of the available heat transfer regimes. For breaks smaller than the two-inch break considered in this analysis, where dependence on steam generator heat removal is increased over that for larger breaks, the homogeneous assumption for the U-tube hot side will alter the time for loss of two-phase natural circulation and condensation heat transfer on the primary side. Analyses to quantify the influence of the homogeneous hot side assumption on small breaks resulting in pressure stabilization above the steam generator secondary pressure or repressurization to the pressurizer PORV setpoint were not performed. Because of the long time intervals at elevated pressures for the very small breaks, the integrated effects on coolant inventory and energy loss through the break will be dependent on such steam generator model assumptions, so that further analyses to resolve this area of uncertainty are recommended. These analyses should also include consideration of hot leg breaks to assess response sensitivity for this type of break.

4.2.1.4 Non-Condensible Gases

In the event of a small break LOCA, non-condensible gases can be introduced into the primary system from various sources. These non-condensible gases can affect the system behavior in a number of ways. Condensation heat transfer in the steam generators can be degraded, non-condensible gas accumulation in system high points can degrade or potentially stop natural circulation flow, and significant amounts of non-condensible gas could introduce errors in those analytical models that are based on equilibrium assumptions.

In Volume 1 of Reference 3, W has provided an evaluation of the effect of noncondensible gases on a small break LOCA. Our review of this evaluation and the conclusions based on our review are provided in the following sections.

4.2.1.4.1 Sources of Non-Condensible Gas

In a PWR, there are nine sources of non-condensible gas which could potentially be introduced into the primary system during a small break LOCA. These are as follows:

(1) Dissolved hydrogen in the primary coolant,
(2) Dissolved nitrogen in the accumulator water,
(3) Dissolved air in the refueling water storage tank,
(4) Hydrogen released from the zirconium-water reaction,
(5) Free nitrogen used to pressurize accumulators,
(6) Hydrogen released from radiolytic decomposition of injected water,
(7) Fission and fill gas in reactor fuel,
(8) Hydrogen gas (free and dissolved) in makeup tank, and
(9) Pressurizer steam space gas.

Westinghouse has accounted for each of these sources, with the exception of the makeup tank gas*, in their analysis of a two-inch cold leg break. The two-inch cold leg break was selected since it is the largest break for which the steam generator is relied upon to remove a significant amount of decay heat. It would also have the potential for introducing the greatest amount of noncondensible gas into the system. In addition, there are no breaks larger than two inches in diameter which could be isolated during the course of the accident. The pressurizer spray line, which is greater than two inches, is automatically isolated when the pressure drops below a preset value. The analysis assumes this isolation valve functions as designed. Emergency procedures are required to prohibit the operator from closing the loop isolation valves.

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*This is a low (~ 35 psig) pressure system. Any rise in tank pressure will result in relief through the tank safety valve.

The W analysis consisted of utilizing system parameters calculated for a two-inch cold leg break to calculate separately, as a function of time, the amount of gas that would be introduced into the system. This approach is valid as long as the amount of gas introduced would have an insignificant effect on the calculated system behavior. This concern is discussed below.

4.2.1.4.2 Effect of Condensation Heat Transfer and Equilibrium Assumptions

During normal liquid film condensation, the condensing vapor flows toward the condensing surface. If non-condensible gas is present, this gas will be carried with the vapor flow toward the condensing surface and accumulate. An equilibrium concentration will be established when the amount of gas being carried toward the surface with the vapor equals the amount of gas flowing away from the surface due to the pressure gradient of the gas.

Westinghouse has provided an analysis of the effect of non-condensible gases on condensation heat transfer utilizing a model which accounts for the processes described above. In this model, the heat transferred across the liquid boundary layer and the steam generator U-tube is equated to the heat transferred by mass diffusion and conduction through the vapor boundary layer. This approach, however, assumes that the heat transferred through the liquid film is equal to the heat transferred through both the liquid and vapor films when non-condensible gases are not present. The heat transfer correlation used is that recommended by Rohsenow and Hartnett.(22) The assumption that the heat transferred through the liquid film is equal to the heat transferred through both the liquid and vapor films is conservative, since the thermal resistance of the liquid film includes that of the vapor as well. In addition, the condensation heat transfer correlation used has not been experimentally verified under conditions and geometries prototypic of small breaks in W-designed steam generators. Comparisons of the W analyses to Semiscale Test S-07-10B (see Section 4.2.1.5 of this appendix) are expected to provide confirmatory information on the acceptability of the condensation heat transfer model.

In order to show the effect of degraded condensation heat transfer due to the noncondensible gas buildup, W performed two small break calculations using the evaluating model version of the WFLASH code.(23) One case was considered a "base case," using nonimal values of the condensation heat transfer coefficient. The second case was a "degraded case," in which the condensation heat transfer coefficient was reduced by 20% to account for noncondensible gas. The 20% reduction is greater than that which would be expected if all of the non-condensible gas calculated to enter the primary system accumulated in the steam generator. Comparison of these results shows that the effect of a 20% reduction in the condensation heat transfer coefficient has a negligible effect on the system behavior, and that the figure of merit, the core mixture level, was essentially unchanged.

Westinghouse has also presented the results of analyses comparing the amount of heat that can be transferred with the expected release of non-condensible gases for a two-inch diameter cold leg break to the amount of heat that could be transferred without non-condensible gases. This comparison showed that, only when the primary system pressure approaches the secondary system pressure does the amount of heat that can be transferred become significantly degraded. The effect would be for the primary system to remain at a slightly higher pressure (25 psi higher, if all the non-condensible gases remained in the steam generators). It is expected, however, that non-condensible gases in the system will collect in system high points, and will not evenly distribute throughout the system (assuming the reactor coolant pumps are tripped). Therefore, the increase in primary system pressure to account for non-condensible gases would be even less than 25 psi. Non-condensible gases mixed with primary system steam will exert a partial pressure, resulting in the primary system pressure being slightly higher than the saturation pressure of the liquid. Westinghouse estimates that this effect is negligible, and would have an insignificant effect on small break behavior. We agree that this effect would be negligible.

4.2.1.4.3 Effect on Fluid Flow

Westinghouse evaluated the case where it was assumed that all the non-condensible gas released for the two-inch break accumulated equally at the tops of the three steam generators for the typical three-loop plant analyzed. Westinghouse reported that the volume of the U-bends is about 70 cubic feet per steam generator or a total of 210 cubic feet. Their analysis showed that the gas calculated to be released would occupy less than 25% of the available U-tube volume. This would have a negligible effect on natural circulation flow, since most of the resistance to natural circulation flow is due to the reactor coolant pumps. Even if the gas distributed evenly and all of it was assumed to accumulate in one steam generator, natural circulation flow in that loop would not be calculated to stop. Even if it did, the other two steam generators would be available for decay heat removal.

During the cooldown portion of the small break LOCA recovery period in which system pressure is decreasing, there is the potential for non-condensible gas which may have accumulated in the steam generator U-bends to expand enough to block natural circulation flow.

Westinghouse guidelines for operator actions have provided instructions for this eventuality, which involve starting the reactor coolant pumps to sweep the gas out of the high points. Even if the pumps were not started, natural circulation would cause the system to pressurize and the core region fluid density to decrease to the point that a sufficient density differential would be established to sweep the bubble out of the U-bends and around the loop.

4.2.1.4.4 Conclusions

We have reviewed the W evaluation of the effect of non-condensible gases on system behavior during small break LOCAs. We have found this evaluation acceptable. However, confirmatory information will be needed in the following areas:

1. Verification of predicted condensation heat transfer rates in the presence of non-condensible gases.

2. Confirmation that the effect of non-condensible gas on system pressure and consequent system behavior is negligible.

4.2.1.5 Experimental Verification

In order to fully understand plant response to small primary coolant system breaks, it is necessary to verify the calculational model used to predict small break behavior.

Many of the individual models within the overall W evaluation model have previously undergone comparisons against experimental data, as well as other methods of verification. However, the TMI-2 accident has emphasized the importance of certain phenomena which are expected to occur during a small break LOCA. From this, the staff has identified certain models, methods, or features of the evaluation codes which require more extensive verification.

In addition to verification of individual models, it is also necessary to assure the proper interaction of these models within the overall systems evaluation model. This is accmplished through verification by comparison to integral systems tests. In the following two sections, both previous and planned integral systems tests designed for small break LOCA code verification are discussed.

4.2.1.5.1 Semiscale Small Break Test S-02-6

As part of both the United States Standard Problem Program and the International Standard Problem Program,* a test was conducted in November 1975 at the Semiscale facility at the INEL in Idaho to simulate the behavior of a 6% small break in the cold leg. This test is designated S-02-6. Comparisons of pretest predictions by PWR vendors with the measured data showed poor agreement. Hwever, certain test measurements were questionable, including the measured break flow and vessel inventory. A more complete description of test S-02-6 and the comparisons of vendor predictions to the measured data can be found in Reference 24.

As a result of this test, the staff concluded that additional data on small break behavior was needed. Available information indicated that large uncertainties in the calculational models could exist and better, more extensive comparisons of calculational models to experimental data were needed.

4.2.1.5.2 Semiscale Small Break Test S-07-10B

As a result of the possible data inaccuracies in Test S-02-6, as well as the inaccurate predictions of the measured data, a second test, designed to provide integral system thermal-hydraulic behavior of a cold leg small break was performed in the Semiscale facility by EG&G Idaho, Inc. on January 19, 1979.

Westinghouse, along with other reactor vendors, performed a "blind"** prediction of this small break test (S-07-10B). The data from this test are being withheld from public disclosure until all of the vendor predictions have been received by the staff. Only the initial test conditions have been provided for the calculations. Westinghouse submitted their pretest prediction of Test S-07-10B on October 26, 1979. The data from Test S-07-10B were released to the public on December 3, 1979 and EG&G, Idaho, will evaluate how well each of the vendor's predictions compared with the test data. From these comparisons, as well as those performed by the staff, the need for each vendor to improve certain models or certain aspects of the models will be determined.

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* The U.S. and International Standard Problem Programs are voluntary participation programs, in which participants predict the thermal-hydraulic behavior of agreed-upon experimental tests. The tests selected are designed to challenge certain methods, features, and/or models of loss-of-coolant accident analysis computer codes used in reactor design and safety evaluations.

**"Blind" predictions are calculations made to predict the expected test behavior. Typically, the test is run and the test results are not released until all of the predictions have been made. The reason the test is run in advance of the predictions is so that the actual initial conditions of the test can be used for the predictions. Once the predictions have been made, then the test data are released.

4.2.1.5.3 LOFT Small Break Tests

In addition to verifying vendor small break analysis models with data from Semiscale Test S-07-10B, a specified small break simulation test (LOFT Test L3-1) has been performed in the LOFT facility. The licensees have been requested to prepare pretest predictions of this test's thermal-hydraulic behavior the same way as was done for Semiscale Test S-07-10B. The specific conditions and characteristics for the LOFT test are as follows:

Power = 50 Mwt.
Heat rate = 16 Kw/ft.
Break size = approximately 2.5%.
Break location = Cold leg.

The test was conducted on November 21, 1979.

In addition to Test L3-1, the staff has required in Reference 4 that all PWR vendors and fuel suppliers provide pretest predictions of LOFT Test L3-6, scheduled to be run in March 1980. LOFT Test L3-6 will be a small break loss-of-coolant test in which the reactor coolant pumps will remain running throughout the test. A more complete description of the basis for this test is provided in Section 4 of Reference 4.

4.2.1.5.4 Testing Basis

As part of the evaluation of the extent of additional small break model verification required in light of TMI-2, the staff concluded that model comparisons to integral system tests in both Semiscale and LOFT test facilities were necessary.* The primary reason for requiring model comparison to tests in both facilities was the need to facilitate extrapolation to full scale.

At present, there are no test data available on small break behavior in a large scale PWR. The data from TMI-2, while extremely valuable, are not complete for the purpose of integral model verification. For example, the actual secondary heat load was not known, as well as the system inventory of PORV discharge mass flow. In addition, the system behavior was primarily applicable to B&W lowered-loop design plants and not W or CE-designed plants.

Because of this, integral systems verification of analytical models and methods must depend on scaled tests. In order to extrapolate the scaled data to full size plants, at least two different size scaled tests are needed. This has been the underlying basis for the NRC's integral systems research program and is accomplished with the Semiscale and LOFT facilities.

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*The staff did not foreclose the option for the industry to propose alternative integral systems tests for model verifcation. However, to date, no alternative tests have been proposed by the industry.

The thermal-hydraulic phenomena that are predicted to occur during a small break LOCA are complex. Because of size and design constraints, neither Semiscale nor LOFT can satisfy all of the scaling requirements needed to directly extrapolate test results to large scale PWRs.

The Semiscale facility has some atypicalities associated with it. For example, the Semiscale facility is highly one-dimensional and, because of this, it has a much larger surface-to-volume ratio than a large PWR. Thus, heat losses from the system are expected to be greater.

LOFT, on the other hand, has a much shorter core (5.5 ft.) compared to a large PWR (12 ft.), although the downcomer height is approximately the same as a PWR. The upper plenum is also disproportionately large.

Thus, while each system has unique atypicalities, they are in many respects complementary, and combined will provide a substantially improved data base for model verification than either facility would on its own.

4.2.1.6 Break Discharge Model

The mass flow rate from postulated reactor system breaks is calculated by WFLASH (23) as the product of the break area and the mass flux (flow rate per unit area). The mass flux is calculated using the modified Zaloudek correlation when the fluid stagnation condition upstream from the break is subcooled. For saturated or two-phase fluid, the mass flux is calculated using the Moody slip flow model.(35) The Moody model was derived theoretically to maximize the predicted flow rate, and the Zaloudek correlation was derived from experimental data and modified by $\underline{W}$ to converge with the Moody model for saturated liquid flow. Investigations by Powell(25), Sozzi and Sutherland(26), Simon (27), and experiments at the Marviken Power Station(28) have demonstrated a wide variation in mass flux as a function of break geometry. Mass flux was shown to be influenced by the degree of curvature at the break inlet, flow passage diameter, flow passage length, and the ratio of the break diameter to the vessel diameter. Correlations incorporating all these factors are not available at the present time. Moreover, small break geometries postulated for reactor systems could range from splits in pipes to double-ended breaks restrained by pipe supports, and could include full ruptures in small diameter pipes.

Westinghouse has provided sensitivity studies on the effect of different break flow models on the amount of core uncovering and peak cladding temperature predicted by WFLASH. These studies included an assumed stuck-open PORV and postulated small breaks in the cold leg. The break flow model was found to have a minimal effect on the core water inventory and the peak cladding temperature. Westinghouse concluded that the

WFLASH small break analytical method, which includes the evaluation of a spectrum of break sizes, is adequate to account for uncertainties in the break flow model produced by variations in break geometry.

The sensitivity studies performed by W involved increasing the flow rate for subcooled and saturated liquid flow above the values predicted by the modified Zaloudek correlation, and using the Moody model unchanged from two-phase and all steam flow. For the case of a stuck-open PORV, the change in the model had little effect, since the break flow was largely two-phase. For the cold leg break case, the flow was subcooled for most of the analysis, so that the increase in the break flow acted like that for a larger break size. These studies covered only a limited number of postulated small break LOCAs and did not include cases which assumed loss of auxiliary feedwater. We will require additional evaluations of break flow in our review of small break models for conformance with Appendix K to 10 CFR Part 50 discussed in Section 4.2.1.10 of this appendix.

Additional insight about the expected break flow and the effect on peak cladding temperatures, and the amount of core uncovering will be obtained from the comparisons of WFLASH predictions with experimental data obtained from the Semiscale and the LOFT facilities, as discussed in Section 4.2.1.5 of this appendix.

To calculate the flow through the PORV, W uses the rated valve for dry steam supplied by the valve manufacturer, in conjunction with the mass flow per unit area predicted by the Moody model. These parameters are used to determine an effective diameter for the PORV. The effective diameter thus determined is used for all flow regimes. PORVs are not tested at rated conditions for either steam or two-phase flow. The Moody model predicts flows for steam that are approximately 10% higher than the Napier formula which is normally used to size PORVs. Comparison with data from the Semiscale facility(29) indicates that the Moody model predicts flows which are 70% higher than the data. For this reason, the WFLASH code's break flow model might overpredict flow rates for the stuck-open PORV case when the discharge flow is two-phase. However, the W model would probably be conservative for the analysis of the fluid inventory lost from the system. For the assumed case of a complete loss of feedwater, it is necessary to open the PORV to depressurize the primary system to allow HPI operation, the WFLASH predictions of the decompression rate may be non-conservative. The staff's requirement for testing the PORV under single and two-phase flow conditions, as recommended in NUREG-0578,(1) should resolve this concern.

4.2.1.7 Vessel Mixture Level

The W evaluation model calculates the mixture level in the vessel using a bubble rise model and a steam separation rate calculated using a drift velocity model. The drift velocity model was correlated to core uncovering tests partially described in Reference 30. The staff has required W to submit a report documenting all of the test data and a comparison of the data to the drift velocity model.

In order to determine its acceptability, the overall model was used to predict the same tests from which the drift velocity model was derived. Comparisons were made of the measured to predicted collapsed liquid level when the measured and calculated mixture levels were equal. These comparisons indicated good agreement at low pressures, but the WFLASH code underpredicted the core liquid inventory for higher pressures and greater amounts of core uncovering.

Westinghouse evaluated the sensitivity of the mixture level model to the drift velocity. A comparison was made between the base calculation with the nominal drift velocity and a calculation in which the drift velocity was multiplied by two (to produce greater phase separation and less level swell). This case resulted in an increase of 69°F in the calculated peak cladding temperature.

Based on the comparisons of the model to experimental data, as well as the sensitivity calculations, we have concluded that the mixture level model in the WFLASH code is adequate for the purposes of developing guidelines for emergency operational procedures. However, we have requested additional information from W in the areas of model development and verification data bases.

4.2.1.8 System Noding Detail

The WFLASH nodal representation was reviewed to determine if sufficient detail is provided to properly account for the flashing of hot fluid at various locations during decompression. A concern in this regard is the flashing of hot fluid following the depletion of the fluid inventory in the pressurizer; the pressurizer normally contains the hottest fluid during steady state operation. Pressure control following this time could be transferred to either the hot legs, upper plenum, or the upper head. Since flow from the upper head is restricted and does not mix with the circulating loop flow, it is potentially the hottest system fluid following emptying of the pressurizer. The WFLASH small break LOCA model does contain a separate node for the upper head, but combines the upper plenum with the core volume. Westinghouse should confirm through nodal sensitivity studies that the small break LOCA model properly accounts for the flashing of fluid which can control system pressure during the decompression process.

Another concern related to system component nodal representation consists of the accumulator injection characteristics resulting from the node size and contents when depressurization to the accumulator setpoint is reached in small break LOCAs. Nodal studies performed by the staff and by the vendors have shown an important accumluator injection dependence on the size of the node being filled by the accumulator. It is believed that such injection characteristics are being overly influenced by the model limitations than by actual component dimensional effects. As a result, nodal size sensitivity studies should be considered in connection with the forthcoming LOFT Small Break Tests to develop nodal modeling criteria designed to produce as near actual accumulator injection characteristics as possible.

4.2.1.9 Equilibrium Assumption in System Representation

All of the major computer codes used by the PWR vendors to predict thermal-hydraulic behavior for small break LOCAs assume thermodynamic equilibrium within a given fluid control volume. This assumption requires that all steam and liquid calculated to exist within a control volume be at the same temperature and pressure. This also requires the assumption of instantaneous mass and energy transfer to maintain equilibrium (for example, superheated steam and subcooled or saturated water would not be allowed to exist within the same control volume).

During the period of recovery from a small break LOCA, the equilibrium assumption requires that the liquid refilling the system condense steam with 100 percent efficiency. If the condensation efficiency was less than 100 percent, then some compression of the steam would occur, raising the system pressure. This, in turn, would serve to increase the break flow and reduce the HPI flow.

Westinghouse has not addressed the non-equilibrium effects during the refilling period of a small break LOCA. We believe that this effect is expected to be small for cases in which feedwater is available. However, there is a potential for the effect to become significant for the case of loss-of-auxiliary feedwater.

While the analyses presented in WCAP-9600(3) are sufficient for developing appropriate operator guidelines, W will have to address these non-equilibrium effects in greater detail as part of the model approved for compliance with Appendix K to 10 CFR Part 50.

4.2.1.10 Conclusions

1. The small break analysis methods used by W are satisfactory for the purpose of performing small break LOCA analyses to develop improved guidelines for emergency operational procedures and the training of reactor operators.

2. Several of the individual models, such as steam generator heat transfer, have been identified in the previous sections as requiring further confirmation. In addition, verification of the total analysis method with appropriate small break integral test data is needed. Experimental small break data will be provided by the Semiscale and LOFT test facilities. Since the models were not developed and reviewed for application to the very small break spectrum, the models should be approved for conformance with Appendix K to 10 CFR Part 50.

Recommendations:

(a) The analysis methods used by W for the analysis of very small break LOCAs to demonstrate conformance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted by W for NRC review. The revisions should account for comparisons with experimental data, including data from the LOFT and Semiscale facilities.

(b) Plant-specific calculations by all licensees using the approved models described in (a) above for very small breaks to show compliance with 10 CFR § 50.46 should be submitted for NRC review.

4.2.1.11 Summary of Requirements for Additional Analyses

From the staff review of Westinghouse's small break LOCA model used in the analyses presented in WCAP-9600, the following is a summary of requirements for additional analyses as discussed in the preceding sections.

1. Pressurizer Model

Additional analyses are required to assess the effect of spray isolation valve failure on pressurizer response in a small break LOCA. Non-equilibrium effects in the pressurizer for such accidents should be evaluated (Section 4.2.1.2.3). Additional analyses of very small pressurizer steam space breaks are required to confirm the absence of conditions that would produce counter-current flow phenomena in the pressurizer surge line (Section 4.2.1.2.6).

2. Steam Generator

Experimental verification of two-phase mixture response and heat transfer in the steam generator U-tubes during condensation heat transfer is required to justify primary side modeling using a bubble rise or homogeneous model. (Section 4.2.1.3)

Additional analyses are recommended to quantify the effect of modeling the steam generator U-tube hot side as a homogeneous mixture on the system pressure response for very small breaks, including hot leg breaks. (Section 4.2.1.3)

3. Non-Condensible Gases

Verification is required to demonstrate that the predicted condensation heat transfer rates in the presence of non-condensible gases are applicable, and that the effects of such gases on system pressure and response are negligible. (Section 4.2.1.4.4)

4. Model Nodal Detail

Additional nodal sensitivity studies are required to confirm the adequacy of the hot region nodal detail to properly account for flashing during depressurization in a small break LOCA. (Section 4.2.1.8)

As part of model verification studies performed using test data from the LOFT facility, the sensitivity of accumulator model response to the size and location of the node attached to the accumulator should be assessed. (Section 4.2.1.8)

4.2.2 Small Break LOCA Calculations

Previous small break analyses performed for plant licensing have been traditionally reviewed by the NRC staff for conformance to regulatory analytical and safety criteria. The staff has not evaluated the adequacy of the analyses as a basis for the preparation of operational procedures, or for the training of operators on emergency procedures. The recent TMI-2 accident has more strongly focused attention on the intimate relationship between these activities. As a result, the NSSS vendors were requested to perform more extensive generic small break LOCA analyses, including scenarios not previously considered or required because they were not expected to result in a greater potential for core damage than those analyses required for licensing. These new analyses were needed to provide a more complete understanding of the progress of slow small break transients as a reference basis for the preparation of operator guidelines, and to assess the required operator response in the event of multiple equipment failures that may lead to core damage. The degree of protection provided by automatic protection systems in the absence of required operator action for these cases was also required to identify any needs for modifications to the automatic protective systems.

Because of the greater number of influential components determining system response to small break LOCAs, the interaction of these components in determining system response presents a more complex set of possible response characteristics than those determined thus far for the large break LOCA spectrum. Such more influential components include the auxiliary feedwater system (AFW), steam generator secondary relief valves, ECCS high pressure injection (HPI) head-flow characteristics and shut-off head, and pressurizer relief valve (PORV) set point. In considering probable system response characteristics over the small break range subject to the actions of some of these plant components, W has identified characteristic break size-dependent behavior modes, limiting breaks in size and location, and small break LOCA scenarios involving operator action not previously analyzed to assess the effects of such action.

4.2.2.1 Small Break LOCA Characteristics

Before proceeding to a quantitative assessment of system response to small breaks to identify the limiting small break size and location, the qualitative aspects of system pressure response to small break LOCAs as a function of size over the small break spectrum were considered. The effect of differences between plants for ECCS pump head design was also included. From these considerations, five different characteristic system responses were identified, and were confirmed by quantitative analyses. Quantitative assessments of small break LOCAs to identify the limiting break in size and location included the analyses of cold and hot leg breaks, pressurizer breaks, a small break in a two-loop plant with one steam generator isolated, and a leak with delayed primary loop pump trip during the LOCA. The loss of feedwater transient coincident with a small break was analyzed to determine required operator action when loss of all automatic feedwater was postulated. The effect of operator isolation of some small breaks during a LOCA was also analyzed.

All analyses performed in the various studies used the October 1975 W Small Break Evaluation Model.(31) Brief discussions summarizing results for each analytical study follow.

The small break LOCA spectrum pressure response was found to demonstrate five different characteristic behavior modes when sensitivity to HPI design is included, which produces differing characteristics for the same size break. These characteristics are shown diagrammatically as approximate functions of break size on Figure VIII-1. With the exception of the smallest break region (0 - 3/8 in. diameter), sample analyses were performed to provide supplementary information for identification of the response modes. The sample analyses utilized four-loop plant designs, and assumed loss of offsite power coincident with scram, so that primary loop flow coastdown to natural circulation occurred for each break. Normal activation of steam generator auxiliary feedwater flow was also assumed.

4.2.2.1.1 Mode 1

Breaks in Region 1 on Figure VIII-1 which can be supplied by charging system input, and which will result in a negligible effect on system pressure will be indicated to the operator in several ways so that normal shutdown and system cooldown can be initiated when they occur. Such breaks can be classed as leaks requiring shutdown under normal operator controlled procedures. No automatic scram, HPI activation, or core uncovering occur, as indicated in Table VIII-6.

In the event of charging system failure when such breaks occur, system depressurization will behave as a Mode 2 break described below. Loss of auxiliary feedwater following such a break would also appear to the operator as a "zero break" loss-of-feedwater transient discussed in Section 4.2.2.3.4 below.

4.2.2.1.2 Mode 2

Breaks in the next size category will produce system depressurization, subsequent reactor scram, and ECCS injection on low system pressure. System pressure stabilization will be established shortly following safety injection, at a level above the secondary side pressure relief setpoint, and be held at this level due to balanced flow rates between safety injection and the subcooled or saturated liquid going through the break. Steam generator heat removal and decay heat transport to the steam generators under natural convection are assumed. Decay heat removal with a fully covered core can be maintained for an extended period until energy lost through the break is equal to energy added by the fission decay process (approximately one day), at which point cooldown will begin with no heat removal through the steam generators. The case of small break LOCAs in this range coincident with loss of feedwater so that steam generator dryout occurs, leading to repressurization and HPI shut-off, is discussed further in Section 4.2.2.3.4 below.

Figure VIII-1

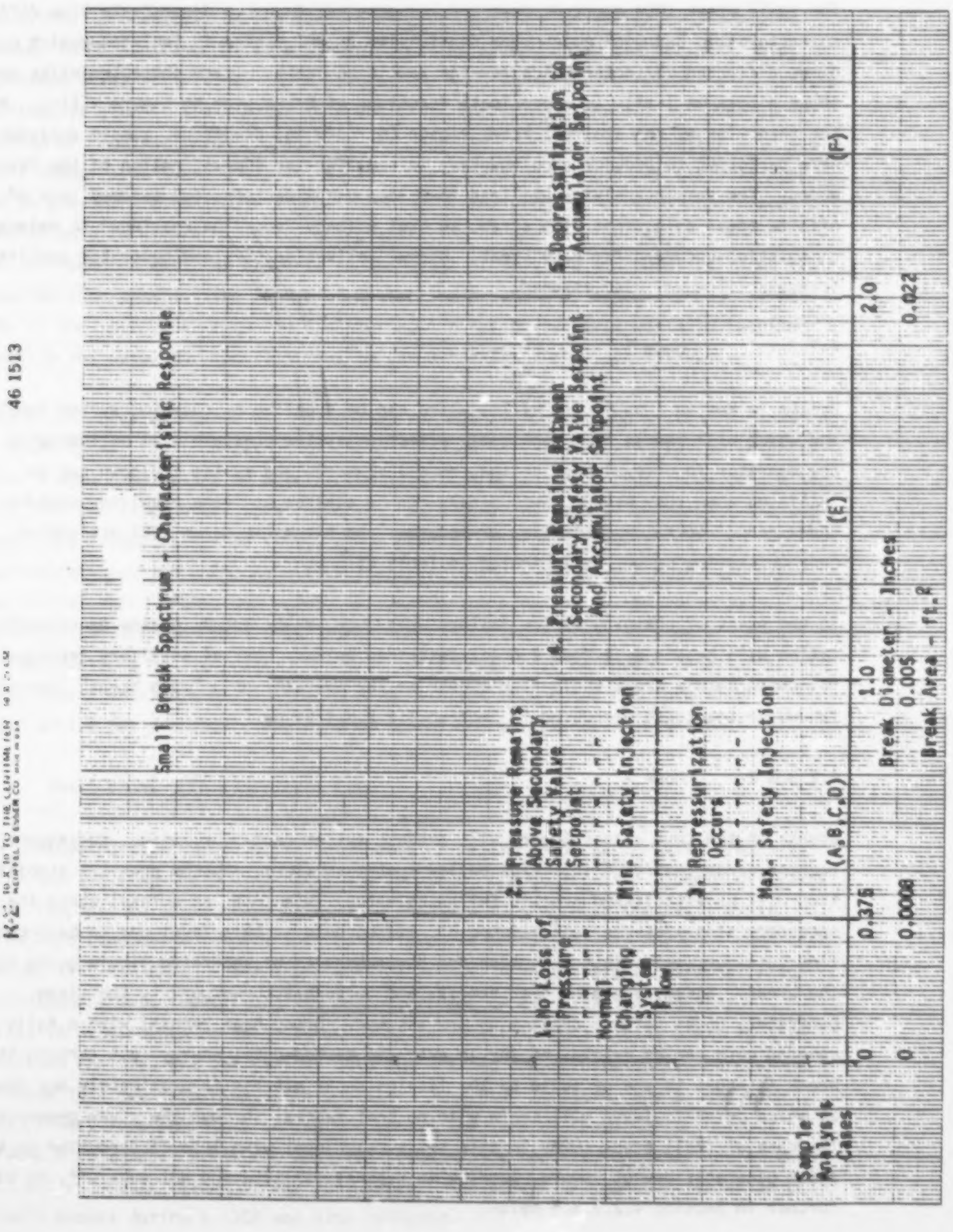

Table VIII-6

SMALL BREAK SPECTRUM BEHAVIOR

Case	Break Range	Size (in. dia.)	Safety Injection (1)	Equil. Pressure (2)	Nat. Circ. Lost (Sec.)	HPI Term.	Core Uncovery	Comments (3)
	<3/8	<3/8	None	Design	Pumps On	No	No	--------
A	$3/8^1$	0.5	Min.	>SG-SV	No	No	No	RESAR-414
B	$3/8^1$	1.0	Min.	>SG-SV	No	No	No	RESAR-3
C	$3/8^1$	0.5	Max.	@PORV	No	No	No	RESAR-3
D	$3/8^1$	0.5	Max.	>SG-SV	No	Yes	No	RESAR-3
E	1^2	2.0	Min.	@SG-SV	400	No	Yes	RESAR-3
F	>2	4.0	Min.	<Acc. SP	100	No	Yes	RESAR-3

Notes:

1. Min. - 1 Charging Pump, 1 HPI Pump, 1 LPI Pump - For plants with safety grade charging pumps. Delete charging pumps for plants with non-safety grade charging pumps.)

   Max. - 2 Charging Pumps, 2 HPI Pumps, 2 LPI Pumps - For plants with safety grade charging pumps. Delete charging pumps for plants with non-safety grade charging pumps.)

2. SG-SV: Steam Generator-Safety Valve
   PORV: Pressurizer-Power operated relief valve
   Acc. SP: Accumulator - Set Pressure level

3. RESAR-414 - No safety-grade charging pumps, HPI shut-off press. = 1500 psia.
   RESAR-3 - Safety-grade charging pumps - HPI shut-off press. = 2100 psia.

From the description of the characteristic response for breaks in this region of the small break spectrum, the pressure stabilization level established after safety injection begins will be dependent on the flow balances established between the break size and safety injection system head-flow characteristics. To demonstrate the effect of safety system design on the pressure response characteristics for such breaks, two analyses were performed and reported in WCAP-9600. In the first of these analyses, Case A, a 0.5-inch diameter cold leg break in a RESAR-414 four-loop plant was assumed, and, in the second analysis, Case B, a one-inch diameter cold leg break was assumed in a RESAR-3 four-loop plant. The Case B plant safety system includes safety-grade charging pumps, compared to the non-safety grade pumps of the Case A plant.

Results of the two analyses demonstrate the anticipated pressure stabilization characteristic, with the pressure settling just under 1500 psia in the Case A plant, and near 1250 psia in the Case B plant. Minimum safeguards safety injection (definition in notes of Table VIII-6) was assumed in both analyses. However, the assumption of maximum safeguard safety injection in Case A would have had little effect on the stabilized pressure level due to the low HPI shut-off head limitations, while in Case B, with higher HPI shut-off head and safety-grade charging pumps, repressurization would be possible.

4.2.2.1.3 Mode 3

In the event of repressurization for breaks of this size, HPI termination may be used to avoid filling of the pressurizer to a water solid condition, with consequent pressurizer relief valve flow of saturated or subcooled liquid at high pressure. Westinghouse HPI termination criteria(14,15,16) will permit operator shut-off of HPI when the system pressure is at 2000 psia and rising, pressurizer level is at or above the no-load range, water level is present in one steam generator for a heat sink, and require full power subcooling, operator to compare system response quantitatively for a given break, with and without the HPI termination, two analyses were performed on the Case B plant of the previous comparison. In this comparison, the maximum safeguards safety injection was assumed so that repressurization would occur. In the first analysis, Case C, HPI was not shut off at any time, while in the next analysis, Case D, an earlier set of W criteria similar to the current criteria outlined above were employed, but without HPI reinitiation that would be permitted by the criteria.

Results of these analyses demonstrate that the core would remain covered in both cases. A solid system with oscillatory pressurizer relief valve operation would be achieved if HPI termination does not occur and, when employing the criteria, HPI could be terminated at about 4560 seconds (approximately 1.25 hours) after the break occurred. Comparison of this termination time with that calculated using the NRC criteria (50°F subcooling in hot leg) specified in IE Bulletin 79-06A has shown that HPI termination could occur at essentially the same time for both criteria for this break size. The extent of this comparability between the termination criteria over the small break range was not investigated further, but is expected to be relatively equivalent over the limited range of the small break spectrum where repressurization by pumped injection occurs. Additional comparisons should be made in view of the revised termination criteria to assess the influence of break size on this feature of the operator guidelines.

4.2.2.1.4 Mode 4

Breaks in this range (one to two inches diameter) of the small break LOCA spectrum will depressurize rapidly to the scram and safety injection actuation setpoints, but will stabilize for an interval at a pressure above the steam generator safety valve set point, as decay heat removal is shared between the break and the steam generator until steam flow discharge from the break occurs. Before steam flow can occur at the break, development of a separated steam volume occurs on the cold side of the steam generator U-tubes, resulting in two-phase mixture drainage from this part of the steam generator with consequent loss of two-phase natural circulation. Steam generator U-tube hot side drainage that would be expected to occur simultaneously is prevented in the model by NRC requirements that a homogeneous mixture model be assumed for this portion of the system. Further model verification of behavior in this component is needed to more realistically simulate coolant phenomena in two-phase transients.

Following steam generator cold side drainage and loss of natural circulation, the reduced heat removal in the steam generator under condensation heat transfer coupled with the coolant boil-off in the core act to raise system hot side pressure slightly to relieve the steam generated by decay heat. The two system manometers represented by the downcomer and core in the reactor vessel, and the pump suction side loop begin to interact in clearing a steam relief path to the break. Reactor coolant system hot side pressure is increased by the mixture elevation differences developed between the downcomer and core, and the resulting pressure gradient serves as the driving head to clear the pump suction side loop seal providing a path for steam relief to the break. Following loop seal clearance, steam flow to the break occurs, resulting in a system pressure decrease, added safety injection HPI flow which recovers the core, and further reduction in coolant steam content and pressure. Heat removal through the steam generator diminishes following loop seal clearance due to the higher HPI flow, and eventually ceases, after which reverse energy transfer from the steam generator secondary side begins.

System pressure gradually decreases below the secondary system safety valve pressure setpoint and stabilizes at an equilibrium pressure determined by break size and HPI head-flow characteristics, at which point long term cooldown can be initiated by the operator.

4.2.2.1.5 Mode 5

Breaks in this range (greater than two inches in diameter) result in more rapid depressurization with a short interval pressure plateau above the secondary side safety valve setpoint before loss of natural circulation occurs. Because breaks in this range can remove the decay heat load independently of the steam generator, rapid depressurization to the accumulator setpoint occurs. Uncovering of the core occurs as before to clear the loop seal. However, the degree of uncovering is to a lower level and over a longer interval due to the greater coolant loss through the break, which exceeds the safety injection rate until pressures are somewhat lower than in the previous mode.

4.2.2.1.6 Summary

From this assessment of general response characteristics for a spectrum of small break LOCAs, it has been noted that the influence of the steam generator is limited to breaks between 3/8 to 2 inches in diameter, and that re-pressurization to a level requiring HPI termination in accordance with the W or the NRC criteria would require maximum safety injection. Only plants having safety-grade charging pumps as part of the safety injection systems would have this capability. No uncovering of the core is predicted for cold leg breaks smaller than two inches in diameter using the single failure criterion required for LOCA analyses by Appendix K to 10 CFR Part 50 for LOCA analyses.

From the analysis performed to identify the small break spectrum response characteristics, it has been shown for Mode 3 conditions that operator action is required to supplement automatic safety systems action to control the small break transient prior to activation of long term cooldown procedures. Multiple failure scenarios discussed later in this appendix will also demonstrate further needs for other supplementary operator action using non-safety grade systems to accomplish the necessary transient control. From these analyses, the need for appropriate operator response during small breaks is identified for use in the preparation of guidelines for emergency procedures and for operator training in coping with small break transients.

The above evaluation indicates that heat transfer to the steam generator via natural circulation is an important heat removal process for break sizes less than two inches in diameter. Natural circulation with subcooled primary system fluid has been satisfactorily demonstrated in many instances. However, two-phase natural circulation has only been demonstrated theoretically and not experimentally. Since two-phase conditions are expected following a small break LOCA, the various modes of two-phase natural circulation should be demonstrated experimentally. The results of the tests should be submitted for NRC staff review. In addition, reactor control rooms should be provided with appropriate means, including additional instrumentation if necessary, to facilitate the determination that natural circulation has been established.

4.2.2.2 Limiting Small Break Analysis

To determine the limiting break size and location for the small break spectrum, W analyzed small breaks in the hot and cold legs of a three-loop plant, because this plant design has previously demonstrated higher peak cladding temperatures in the analyses of small break LOCAs than similar analyses for two and four-loop plants.

All analyses assumed loss of offsite power at scram, and normal auxiliary feedwater actuation following scram. Because of the potential for greater inventory loss for small break LOCAs with full primary flow, particularly for hot leg breaks, the influence of the loss of offsite power assumption has been analyzed more extensively in a separate study.(32) The staff evaluation of that study will be reported separately. For the limiting case analyses, all Appendix K requirements were imposed in the same manner used in a safety analysis report. The series of small break LOCAs analyzed to identify the most severe, or limiting break in terms of highest peak cladding temperature (PCT), is listed on Table VIII-7. The first eight cases (Cases A-H) constitute the analyses used to identify the limiting break in size and location. The next two cases (Cases I and J) considered the limiting break in more detail as a function of minimum and maximum safeguards safety injection for a three-loop plant. Case K is identical to Case B, consisting of an extension of the transient out to 20,000 seconds (5.5 hours), to demonstrate stable cooling conditions. For greater generality, bounding minimum and maximum safety injection curves were developed representing the composite high head safety injection flow/MWt as a function of pressure for all W-designed plants. Using these bounding curves, the effect on PCT for the limiting break and breaks on either side of the limiting break cases (Cases L-O) was also assessed.

Results from the limiting small break LOCA analyses are summarized graphically on Figure VIII-2 showing the relative PCT distribution for the various conditions analyzed in the study. A brief discussion of each phase of the study follows.

4.2.2.2.1 Limiting Break Analysis

To identify the limiting small break, eight breaks were analyzed; four cold leg pump discharge breaks (Cases A through D), a cold leg pump suction break (Case E), and three hot leg breaks (Cases F through H). Results summarized in Table VIII-7 for cases A to H show that the highest PCT predicted for small breaks occurs for the three-inch diameter break in the cold leg pump discharge with a peak cladding temperature of 1708°F. All cold leg breaks in this spectrum demonstrated uncovering of the core as a result of the rapid coolant loss and the pressure gradient required to clear the loop seal, as described in Section 4.2.2.1.4 above. No uncovering of the core was computed for the hot leg breaks, due to the greater energy removal through the break somewhat earlier in the blowdown transient for these breaks.

To assess the PCT sensitivity to the conservatisms of the assumptions on safety injection spillage in the broken loop required by Appendix K to 10 CFR Part 50, two analyses were performed at the limiting break size the first (three-inch diameter) assumed no spillage with minimum safety injection flow (Case I), and the second assumed spillage but included maximum safety injection (Case J). The results are shown in Table VIII-7 and Figure VIII-2. The results indicate essentially identical PCT results (944°F and 928°F, respectively), well below (by about 750°F) the Appendix K calculation.

TABLE VIII-7

LIMITING SMALL BREAK ANALYSES

Case	Break Size (In.-Dia.)	Location (*)	Reactor Scram (Sec.)	Safety Injection (Sec.)	Accumulator Injection (Sec.)	PCT (°)	PCT Time (Min.)
A	2	CL	56.4	65.0	None	1003	25.0
B	3	CL	26.0	32.3	1315	1708	23.7
C	4	CL	17.1	22.0	670	1541	11.4
D	6	CL	11.4	14.3	265	1004	4.8
E	3	PS	22.6	31.8	1433	964	10.4
F	2	HL	104.8	112.3	None	**	**
G	3	HL	41.2	48.3	1482	**	**
H	6	HL	11.4	14.3	265	**	**
I	3	CL	26.0	32.3	1394	944	9.1
J	3	CL	26.0	32.3	1333	928	9.2
K	3	CL	26.0	32.3	1315	1708	23.7
L	3	CL	26.0	32.3	1233	2169	21.1
M	2	CL	56.0	62.9	None	1089	23.8
N	3	CL	26.0	32.3	1291	949	9.0
O	4	CL	17.1	22.5	698	957	4.6

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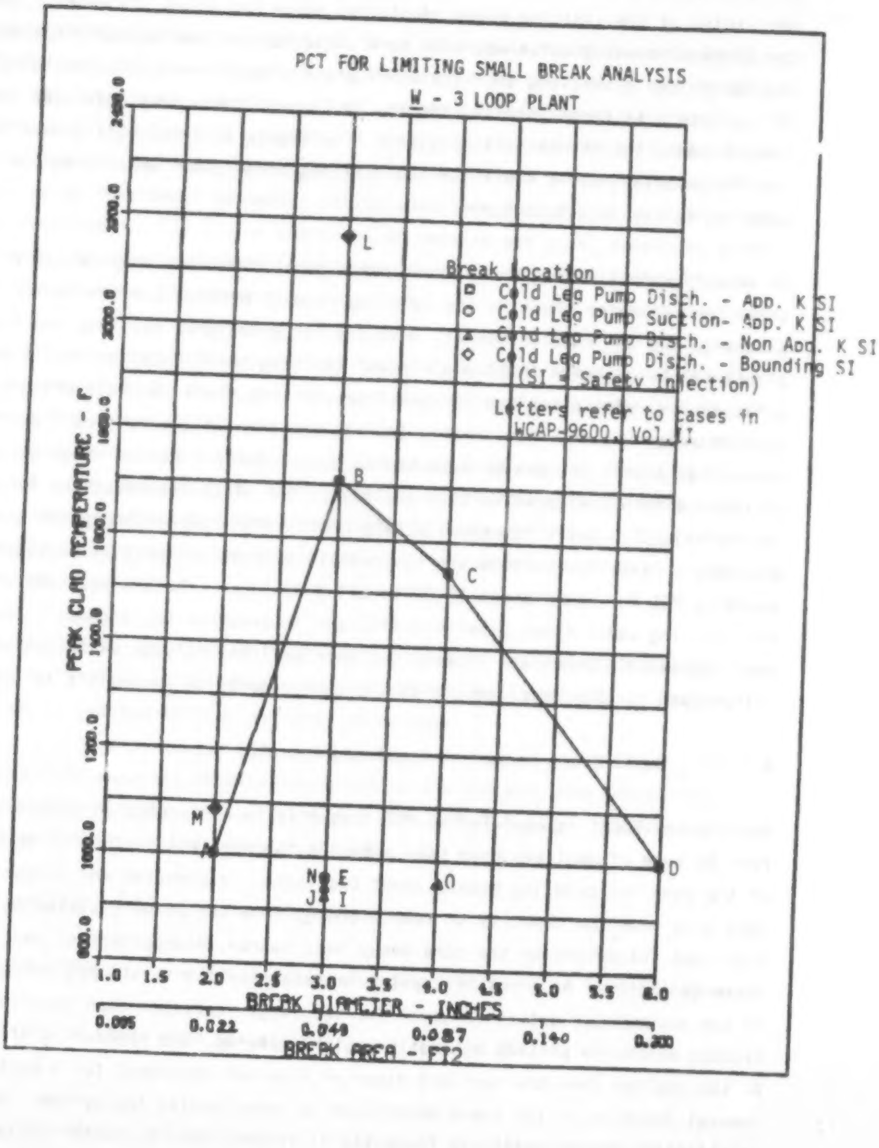
* CL - Cold Leg-Pump Discharge
PS - Cold Leg-Pump Suction
HL - Hot Leg
** - No Heatup

Figure VIII-2

FIGURE VIII-2



To provide greater generality in the limiting break study by including consideration of all W plant designs, composite minimum and maximum safety injection curves were developed, based on the minimum safeguards safety injection flow rates/MWt versus pressure for all W-designed operating plants. The resulting curves constitute bounding minimum and maximum curves for the minimum safeguards for all W plants (regardless of loop configuration), and can represent more conservative safety injection assumptions for some plants than those required by Appendix K to 10 CFR Part 50. Applying the bounding minimum and maximum safety injection flow rates to the three-loop plant in the region of the limiting break identified above has shown, in Cases L and M, that the minimum bounding curve would be more conservative than Appendix K assumptions for the three-loop plant, and that the maximum curve would produce essentially identical PCT results with those obtained for the non-Appendix K safety injection assumptions of Cases I and J for the three-loop plant. The degree of conservatism over the Appendix K results amounts to over 400°F for the limiting break case, when using the minimum bounding safety injection curve.

In summary, the limiting small break for a generic W three-loop design will be a three-inch diameter break in the cold leg segment between the reactor coolant pump discharge and the reactor vessel. Analyses for W-designed two-loop and four-loop plants can be expected to show a similar limiting break location, but with lower PCTs. Because of varying plant coolant inventory distributions, and differing safety injection characteristics, the limiting small break size will vary from that found for the three-loop plant, but can be expected to occur within a narrow range of the spectrum of break sizes considered in this analysis. Use of the conservative minimum composite safety injection curve for the limiting break in the three-loop plant analysis (Case L) provides a basis for considering the results of that analysis as representative of the bounding PCT for small break LOCAs in all W designs. The analyses performed to identify the limiting small break provide sufficient information for a generic understanding of small break LOCA behavior. However, plant-specific analyses will continue to be relied upon to show compliance with the requirements of Appendix K to 10 CFR Part 50.

4.2.2.3 Small Break Scenarios Requiring Operator Action

Small break LOCAs accompanied by the system failures assumed in Appendix K to 10 CFR Part 50 type of analyses have been shown in the previous section to result in uncovering of the core for cold leg breaks about two inches in diameter and larger. Breaks of this size have the capacity to remove energy from the primary system at a rate greater than that introduced by the core decay heat source, independent of heat removal by the steam generator. As a result, system depressurization to the HPI actuation level, or to the accumulator actuation level for the larger (greater than four inches in diameter) breaks, occurs to provide automatic coolant make-up, and recovering of the core. Breaks smaller than the two-inch diameter size are dependent for a portion of the heat removal function on the steam generators to depressurize the system. Appendix K-type calculations assume auxiliary feedwater is present when a reactor system shutdown is initiated. Operational experience has shown, on the other hand, that failures of the main and auxiliary feedwater system to respond as designed have occurred, so that failure to obtain automatic protective action for the small breaks must be considered and operator action may be required to bring the system into a long-term stable condition, from which long-term cooldown can proceed. Operational experience has also shown that relief valves can fail to reseat properly when pressures recede below the setpoint following valve lifting, and, in the case of a pressurizer relief valve, could serve as a small break LOCA coincident with a loss-of-feedwater event, such as was experienced at TMI-2.

Previous licensing analyses have assumed that these components function as designed since their individual failures did not result in more severe uncovering of the core transients than the transients reported. It was assumed that appropriate operator action was taken when such failures occured. A more realistic approach is considered to be one which attempts to bound the probable multiple failure scenarios, and relies on operator action to supplement automatic action when such failures occur. This has led to updated requirements for closer coordination between analyses, operating guidelines and operator training in recognizing system generic response to such accidents and the operator procedures for recovering from them.

To evaluate required operator action for those types of accident scenarios not fully controlled by automatic safety grade systems, performed an accident study (Volume III of WCAP-9600) covering a variety of possible small break scenarios in the break size region (0-two inches in diameter) where the system decay heat removal function is shared between the break and the steam generators to depressurize the system to the ECCS setpoints. These analyses are summarized briefly in Table VIII-8. This table includes as the first eight cases the results of the small break spectrum characteristic responses discussed in Section 4.2.2.1, above, for comparison purposes.

From this table, in the column labeled "operator-action," it is seen that operator intervention for the variety of accident scenarios considered in the "Syst. Failure Assumption" column is limited to four actions, as follows:

(1) Termination of HPI when all termination criteria are present, and throttling of HPI flow to maintain system pressure between 1800 and 2000 psia, in accordance with the guidelines presented in Reference 14.

(2) Initiating AFW flow if AFW flow is not present.

(3) Opening of the PORVs as an optional action for the conditions in Item (2) above.

(4) Isolation of breaks where such means exist; such as, the pressurizer PORVs and letdown line.

Maximum allowable time delay for operator action to avoid or minimize uncovering of the core is shown in the "operator - when" column, and the principal plant parameters that will indicate the nature of the transient in progress to the operator are shown under the column headed "Operator Indication." The characteristic system pressure response asymptotes are indicated in the "Pressure Response" column, and the actuation time for HPI and AFW for the various cases are shown in their respective columns. Interruption of natural circulation based on loss of mixture level in the steam generator U-tubes on the cold side is shown to occur for all small breaks (less than two inches in diameter) with loss of auxiliary feedwater, as well as for the larger breaks (greater than or equal to two inches in diameter) with AFW present, and for the breaks resulting from stuck-open PORVs. Core uncovering, where predicted to occur for these cases, is not extensive. The PCT remains below levels specified by the criteria of 10 CFR 50.46 and below the highest PCT predicted in the limiting break analyses of section 4.2.2.2 for the three-inch diameter break in a three-loop plant.

Further discussion of each of the small break scenarios for the last 12 cases shown on Table VIII-8 is presented in the following subsections.

4.2.2.3.1 Pressurizer Breaks

To consider reactor coolant system (RCS) hot region breaks other than in the hot leg, as in Section 4.2.2.2 above, W considered pressurizer vapor space breaks equal in size to approximately one of the smallest pressurizer relief valves (1.35 in^2), and three of the largest relief valves (4.91 in^2) used in W-designed plants. These two breaks bound the range of possible pressurizer breaks, since some plant designs contain three pressurizer relief valves in this size range. The analyses conducted on a four-loop plant for these breaks assumed Appendix K to 10 CFR Part 50 requirements with loss of offsite power on scram, and normal actuation of auxiliary feedwater as a result of main feedwater trip on scram.

(1) Case I

For the smaller break, an initial loss of pressurizer mixture level occurs due to steam flow through the break before the pressurizer fills with a two-phase mixture originating from flashing in the pressurizer and two-phase flow from the hot leg. Flashing and steam separation in the reactor vessel result in a vessel mixture level reduction down to the level of the hot leg nozzles in the upper plenum. Because of the slow depressurization, the coolant in the upper plenum is essentially at saturated conditions, and remains in this state so throughout the transient. Vessel level is held at the hot leg elevation by the combined effects of inventory and energy removal through the relief valve, decay heat energy removal through the steam generator, and inventory make-up from the safety injection system.

Phase separation in the cold side of the steam generator U-tubes results in a reduction in mixture level and loss of single-phase natural circulation starting about 5500 seconds (1 hour, 32 minutes) after the break occurs, and stabilizes near 12,500 seconds (3 hours, 28 minutes) at the 20-foot elevation, at which point long-term equilibrium conditions are established. In this condition, the

W Small Break Analyses

(A) Break Mode	Size	(B) Syst. Failure Assumption	Operator Indication	(C) Pressure Response	(D) HPI On	(D) AFW On	(D) Nat. Circ. Interrupt	(D) Core Uncov.	(E) (ft) Depth	(°F) PCT	(D) Operator Action	When	(D) Long Term Stab. Conds.	Comments
1	0-3/8" (CL)	Appen. K	Vol. Control Charging Flow Contain. Rad	$P \geq P_{DES}$	No	No	No	No	-	-	[illegible]	Low Vol. Contr. Lev.		
1	0-3/8" (CL)	Appen. K + Charging Pump	Lo Press. Scram + HPI Act.	$P > P_{SEC}$	>4000	~4000	"	"	-	-	Oper Eval	-	10,000	
2	0.5" (CL)	Appen. K	"	"	~3500	~3600	"	"	-	-	"	-		
2	1.0" (CL)	"	"	"	~450	~500	"	"	-	-	"	-		HPI Shutoff 1500 psia
3	0.5" (CL)	Max. HPI (4 Loops - Charg)	"	$P = P_{PORV}$ = 2350 PSIA (2 PORVs)	~3100	~3200	"	"	-	-	"	-		HPI Shutoff 2100 psia
3	0.5" (CL)	"	"	P = 2000	~3100	~3200	"	"	-	-	HPI Termination	4650		No HPI Spillage
4	2.0" (CL)	Appen. K	"	$P \approx P_{SEC}$ S.V. Setp.	~120	~150	~400	~140	6.5	<1200	Oper Eval	-		HPI Throttle Below 2000 psi
5	4.0" (CL)	"	"	$P < P_{ACC}$	~105	~140	~160	~450	4.9	<1800	"	-		$P \approx P_{SEC}$
-	1.38" (1 PORV)	"	"	$P \approx P_{SEC}$	~130	~165	~5500	No	-	-	-	-	12,000	$P < P_{ACC}$
-	2.5" (3 PORVs)	"	"	$P \approx P_{ACC}$	~36	~70	~560	"	-	-	"	-	2600	
5	4.0" (CL)	Appen. K - (Pump Trip Delay)	"	$P < P_{ACC}$	~105	~140	~160	~470	4.9	<1800	"	-	>3000	RCP Trip at 1250 psia
	0"	Appen. K + LOFW + LAFW	S.G. SEC Level Press. Level Hi Press	$P = P_{PORV}$	4635	No	~2650	<100	11.0	<1000	Initiate AFW	4000	~10,000	

Table VIII-8

Table VIII-8 (Continued)

(A) Break Mode	(A) Break Size	(B) Syst Failure Assumption	Operator Indication	(C) Pressure Response	(D) HPI On	(D) AFW On	(D) Nat. Circ Interrupt	(D) Core Uncov.	(E) (ft) Depth	(°F) PCT	(D) Operator Action	(D) Operator When	(D) Long Term Stab. Conds.	Comments
-	0"	"	"	"	"	"	2550	105 1100	9.0" 10 3		Open PORV	2500		
1	0.214" (CL)	"	"	"	4034	"	2475	None	-	-	Initiate AFW	3500	17,000	HPI Term. at 10,000 Secs
2	0.420" (CL)	"	"	"	<4000	"	~2420	"	-	-	"	3350	>10,000	HPI Shutoff 1500 psia
4	1.9" (CL)	Appen. K	Lo. Press. + HPI Act.	$P > P_{SEC}$	211	230	1141	80	~0.6	871	Oper. Eval.	-	7000	All Steam Gens. Avail.
4	1.5" (CL)	"	"	$P > P_{SEC}$	211	230	1150	155	~0.5	864	"	-	7000	1 Steam Gen. Isolated
-	2.5" (PORVc)	Appen. K + LOFW + LAFW	"	$P \leq P_{SEC}$	~80	No	~400	None	-	-	PORVs Isolated	<400	8000	
-	"	"	"	"	~80	"	~400	"	-	-	Activate AFW	4000	8000	
4	2.0" Ltdn. L.	"	"	"	~120	"	~80	100	~5.2	>1600	Letdown Line Isol.	1900	3000	

(A) CL - Cold Leg Break
PORV - Pressurizer Relief Valve Stuck Open
LTDN. L. - Letdown Line Break

(B) LOFW - Loss of Main Feedwater
LAFW - Loss of Auxiliary Feedwater

(C) P_{DES} - Design Operating Pressure
P_{SEC} - Steam Generator Secondary Safety Valve Setpoint Pressure
P_{PORV} - Pressurizer Relief Valve Setpoint Pressure
P_{ACC} - Accumulator Setpoint Pressure

(D) Time After Break - Seconds

(E) Mixture height above bottom of core

* Two uncovery interactions at top of core

best copy available

reactor vessel remains filled to the hot leg elevation, and the cold side of the RCS will remain full of liquid and near saturation. Reduced decay heat generation will eventually result in a refilling of the steam generator U-tubes on the cold side with mixture due to reduced heat removal requirements, and RCS pressure will remain pegged to the secondary side pressure until all decay heat removal can take place through the relief valve.

(2) Case II

For the larger pressurizer vapor space break, depressurization is somewhat faster, and no level reduction occurs in the pressurizer before it fills with two-phase mixture. Because of the greater heat removal capacity of the larger break, less heat removal through the steam generator is required for the break, and system pressure decay is controlled by the steam generator secondary side for a much shorter interval. From the computed results, system pressure is reduced below the secondary side relief valve setpoint at about 1000 seconds (17 minutes), indicating the point at which all decay heat removal is taking place through the break. Pressure reduction continues to a point where an equilibrium pressure condition is established between leak and injection rates (about 700 psia) at about 2500 seconds (42 minutes), from which point a slow depressurization to the accumulator setpoint will occur under control of the decay heat generation rate. Vessel mixture height remains near the hot leg nozzles in the upper plenum throughout this transient, but the vessel mixture does display a small amount of quality as a result of the more rapid depressurization.

For each of these bounding cases for pressurizer vapor space breaks, no core uncovering is predicted. However, for the smaller breaks, decay heat removal is dependent on the presence of AFW in the steam generators to avoid repressurization and mixture level reduction before long-term cooling measures can be started. For the larger break, the presence of heat removal through the steam generators is not required, due to the energy removal capability of the break. From these results, the pressurizer vapor space breaks are shown to demonstrate a break size dependence on steam generator heat removal capability similar to tjat exhibited by cold leg breaks.

4.2.2.3.2 HPI Termination

HPI termination for a Mode 3 type break was found to be necessary to limit repressurization for a 0.5-inch diameter break, as discussed in Section 4.2.2.1.3, above. Comparison between the application of the W and the NRC termination criteria for this case would have resulted in essentially identical timing for HPI termination. Further consideration of the break size spectrum over which HPI termination would be needed found that breaks on the order of one-inch diameter were an upper bound on breaks that could produce the conditions allowing HPI termination using conservative break discharge characteristics. Best estimate discharge coefficients would increase this bounding value to about 1.5 inches. The bounding values apply to breaks in the cold legs. Further consideration of PORV failures in a stuck-open position have shown that the resulting transient would not result in conditions satisfying the termination criteria. Breaks with areas smaller than that represented by a stuck-open PORV valve using best estimate discharge models would be required to produce the necessary conditions allowing HPI termination.

One additional accident involving the conditions allowing HPI termination is described in Section 4.2.2.6 below, and in this case, as in the previous case for the small break of Section 4.2.2.1.3, HPI termination under the W or the NRC criteria would occur at essentially the same time.

4.2.2.3.3 Reactor Coolant Pump Trip During Small Break LOCA

Most large break LOCA analyses to date have concluded that tripping of the reactor coolant pumps at the initiation of the accident leads to the highest calculated peak cladding temperatures. Assuming pump trip at the start of the accident was also consistent with the assumption that offsite power was lost. Because this proved to be typically the most limiting conditions for the reactor coolant pumps during the large break LOCA, it was generally assumed that it produced the most limiting conditions as well for the small break LOCA.

Subsequent to the TMI-2 accident, W, along with the other two PWR vendors, performed analyses of small break LOCAs, in which the pumps were assumed to remain running during the accident. They concluded that the major effect of pump operation during a small break LOCA was to redistribute liquid in the primary system. This redistribution resulted in the liquid being made available to the break for a longer period of time than for the same case with the pumps tripped. They concluded that, for a given range of small break sizes, there existed a range of assumed times for pump trip (e.g., mechanical failure, operator action, etc.,) such that the resulting depth and duration of core uncovering would result in cladding temperatures in excess of the 2200°F licensing limit.

We have performed a generic assessment of the effect of delayed reactor coolant pump trip. This assessment and the resulting conclusions are documented in Reference 4. The key conclusions of Reference 4 are: (1) the uncertainty in small break LOCA analysis models with pumps running is large, and cannot at this time be quantitatively relied upon, (2) the reactor coolant pumps need to be tripped early in the accident, and this trip should be performed automatically, and (3) that model verification by pretest prediction of a forthcoming LOFT small break test with the reactor coolant pumps running will be required.

4.2.2.3.4 Operator Response for Loss of Feedwater

In determining appropriate operator response in the event of an extended (main and auxiliary) loss of feedwater flow and a loss of feedwater accident coincident with a small break LOCA, W has assumed that operator response must be adequate to remain in compliance with 10 CFR §50.46 and Appendix K to 10 CFR Part 50.

Loss of main feedwater flow events with failure to obtain auxiliary feedwater flow will result in primary pressure increase to the pressurizer power-operated relief valve (PORV) setpoint. Normal valve operation would maintain system pressure near the valve setpoint, and would require operator intervention to depressurize the system by manually holding the valve open.

An extended loss of feedwater event coincident with a small cold leg break would aggravate the inventory loss if the break size is in the range requiring heat removal through the steam generators to avoid system repressurization. With pressurization to the PORV setpoint, two paths for decay heat removal and inventory loss will exist, and considerable inventory loss could result if timely operator action to reduce pressure and activate safety injection is not initiated.

To obtain a conservative operator response time criterion for all W-designed plants for such accidents, W performed analyses on a two-loop design, because of the smaller coolant inventory relative to the valve and break sizes applicable for these plants. Four cases were considered to provide bounding results for the two accident scenarios described above. The first of these was the extended loss of feedwater accident with a normally functioning PORV, and was labeled as the "zero break" case. From a sensitivity study, the largest coincident cold leg small break that would not result in depressurization to the safety injection setpoint was determined, and labeled the "minimum break" case. Two possible operator actions were then assessed for the "zero break" case. In the first of these actions, the operator manually opened all PORVs to the full open position at 2500 seconds (42 minutes). In the other action, the operator initiated auxiliary feedwater flow at 4000 seconds (67 minutes). The two cases were analyzed with the results summarized in Table VIII-9.

For the "zero break" case defined above, steam generator dryout occurs in 1900 seconds (32 minutes), and repressurization to the PORV setpoint with all decay heat removal through the PORV follows in a little over three minutes. Core uncovering begins in 4035 seconds (67 minutes), and is completed in 16 minutes if no operator action occurs.

For the case where operator intervention occurs at 4000 seconds by means of auxiliary feedwater (AFW) activation (Case A on Table VIII-9), a small amount of core uncovering (approximately one foot) will occur before AFW cooling reverses the coolant inventory loss. System pressure is reduced rapidly, resulting in closure of the PORV and initiation of safety injection flow about 10.5 minutes after the start of AFW. System cooling and refilling will proceed to long-term cooling with no further core uncovering.

If operator intervention is by means of opening the PORVs at 2500 seconds (Case B), more coolant inventory is lost, and a greater amount of core uncovering occurs due to the decay heat removal through the PORVs. Safety injection actuation occurs 13.5 minutes after opening of the PORVs. Reactor vessel mixture level is near the hot leg elevation at this time, and is decreasing due to a continuing inventory loss, even with safety injection. Core uncovering begins about 10 minutes after safety injection starts. The core uncovers a little over two feet below the top of the core. At this point, refill begins and recovers the core in just over two minutes, as a result of loop seal clearance, which occurs at this time.

Following loop seal clearance, vessel mixture level is reduced a second time due to the continued inventory loss, and uncovering of the core to about 1.7 feet below the top of the core occurs 30 minutes after safety injection has started. System depressurization reaches the accumulator setpoint two minutes later, and the core is recovered in just over six minutes after the start of uncovering. Following accumulator injection, pumped safety injection will establish the long-term decay heat removal mode.

For the "minimum break" accident, or the extended loss of feedwater coincident with a small cold leg break, the same operator intervention was analyzed. The largest break that would not result in depressurization to the safety injection setpoint (1760 psia) for this accident was found from a sensitivity study to be a 0.00025 ft^2 (0.214 inch-diameter) break. As a result of this small size, the extended loss of feedwater accident accompanied by this break and without operator intervention is essentially identical to that described above for the "zero break" case. Steam generator dryout occurs in 32 minutes, and uncovering of the core would begin in about 58 minutes, or just less than nine minutes earlier than for the "zero break" case as a result of added inventory loss through the break. Complete uncovering of the core would follow in a slighty shorter time interval than that found for the "zero break" case, if operator intervention does not occur.

Operator action activating AFW at 58 minutes in the transient will prevent core uncovering by cooling of the steam generator primary side coolant, with a consequent reduction in system pressure. This pressure reduction produces a vessel mixture level swell that keeps the core covered, and is followed within two minutes by closure of the PORV due to pressure reduction below the valve setpoint. Pressure reduction continues down to the safety injection setpoint due to the steam generator heat removal capacity, and safety injection begins about nine minutes after the start of AFW. Further cooling and depressurization continue, with the primary system refilling until the accumulator setpoint pressure is reached 40 minutes after AFW initiation. Continued cooling eventually reduces heat removal at the steam generator to a low level, and depressurization is reversed due to continued safety injection. The subsequent pressure rise results in the presence of all conditions required for HPI termination by the W̲ criteria-about 1 hour and 40 minutes after AFW activation. The NRC 50°F subcooling criterion for HPI termination would have been satisfied about two minutes earlier.

Table VIII-9

LOSS-OF-FEEDWATER RESPONSE

	Accident	Operator Actions		HPI Activation (Secs.)	Core Uncovering Time-Secs.	Core Uncovering Depth - Ft.	Core Recovering Hrs.-Secs.
		Comp.	Time-Secs.				
A	LOFW-Zero Break	AFW	4000	4635	4000	1.0	4100
B	LOFW-Zero Break	PORV	2500	3286	3915*	1.7	5600
C	LOFW + Min.** Break	AFW	3500	4009	None	None	None
D	LOFW + Min.** Break	AFW	3500	3375	None	None	None

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*A short uncovering interval preceeds a longer interval. Times shown included both uncovering intervals.
**System with SI system shut-off head of 1500 psia.

For plants having safety injection systems with a 1500 psi shut-off limit, a larger coincident cold leg break would be sustained without depressurization to the safety injection actuation setpoint. To assess this case, the largest break providing this condition, a 0.001 ft^2 (0.43 inch-diameter) break, was analyzed (Case D). Because of the larger break size, repressurization to the PORV setpoint occurs at 2300 sec. (38 minutes), or just over three minutes later than in Case C above. The larger break also results in a slightly shorter interval to core uncovering without operator intervention, and reaches the start of core uncovering about two minutes earlier than in Case C for the smaller break. With operator intervention, as in Case C at 58 minutes, the subsequent transient would be essentially identical.

From the results obtained in these analyses, W has concluded that the most effective operator intervention for the extended loss of feedwater accident is the activation of the AFW within 4000 seconds (67 mins.), and that, failing the availability of AFW at 2500 seconds (42 mins.), the PORVs can be manually opened to provide adequate core cooling for the accident transient. In the event of coincident small breaks which do not depressurize the system to the safety injection level, the available operator time for AFW actuation is reduced from the 67 minutes for extended loss of feedwater alone to 58 minutes for systems with high safety injection setpoints (1760 psia), and to 56 minutes for systems with 1500 psia shutoff limits on safety injection.

The W analyses based on opening two PORVs assumed valve flow rates based on limited experimental data. The effective valve area was determined by applying the Moody model for steam to test data for a typical valve at low pressure with steam. The discharge valve area determined by this method was used with the Moody model to obtain the valve flow with two-phase fluid. The Moody model for steam flow is approximately a best estimate representation. However, the Moody model overestimates flow in the two-phase region in some cases by 70%. In addition, the uncertainty for two-phase critical flow models versus experimental data is ±40%. Therefore, it is likely that the W method overestimates the PORV flow, and that the valve capacity is not sufficient to depressurize the primary system sufficiently to permit adequate core cooling through HPI assuming loss of all feedwater. The importance of PORV discharge rates is recognized in NUREG-0578,(1) Section 2.1.2, which requires that prototypical tests with subcooled and two-phase fluids be performed by July 1981.

4.2.2.3.5 Isolated Steam Generator

To assess the influence of steam generator isolation on a small break transient, a two-loop plant was analyzed to evaluate the case of minimum heat removal capacity for such a combined system failure. Results for a two-loop plant would bound the effects that would be found for three- and four-loop plants for a single steam generator isolation. Steam generator isolation for this analysis consisted of failure to initiate auxiliary feedwater flow after scram and normal secondary safety valve operation at setpoint pressure.

A 0.01 ft^2 (1.5-inch diameter) cold leg break was used for this comparison. It was found that very similar transients would occur with auxiliary feedwater to one, or both, steam generators in a two-loop plant. Because of the water inventory in the steam generator secondaries, loss of natural circulation during the blowdown transients occur at essentially identical times (1141 and 1150 seconds) in the two loops, and the steam generator U-tubes are covered in both steam generators at this time for both steam generator operational configurations. Because of the division of heat removal between the break and the steam generators, the loss of secondary inventory in the isolated steam generator is quite slow, so that appreciable heat removal continues in the isolated steam generator well beyond the time when recovering of the core has been effected by the safety injection system. Uncovering of the core to the 8.5-foot level occurs at about 1575 seconds (26 mins.) into the transients for both cases, with an 88-second interval of uncovering for the core with both steam generators, and a 92-second interval for the isolated steam generator case.

Results presented for a plant with a low shutoff head safety injection system also demonstrated similar behavior, with similar core uncovering at 1700 seconds (28 mins.) over a 107-second interval. For three- and four-loop plants, where a smaller proportion of decay heat removal would occur in the isolated steam generator, less core uncovering and uncovering time would be expected. As a result, it is concluded that isolation of one steam generator during small break transients has a negligible influence on the small break transient.

4.2.2.3.6 Break Isolation

Operator isolation of some small primary system breaks is possible, and the time during transients at which it would be most inopportune to do so have been considered by W. Such times would be:

(1) When repressurization begins for small breaks.

(2) When heat transfer capability is lost in the steam generators due to primary side drainage.

(3) When the primary system pressure is below the secondary system pressure.

(4) When minimum system inventory exists during a transient.

From the analyses of several transients, it was found that the fourth criterion essentially encompassed criteria (2) and (3), and that criterion (1) would occur after break isolation. The last criterion would be the most likely to result in uncovering of the core, as minimum inventory will exist when pump injection rate equals the break flow, and break isolation will act to reduce mixture level by bubble collapse and safety injection reduction as pressure increases.

To assess system response with operator isolation of the break, a four-loop plant having a 0.034 ft^2 break in the pressurizer with the assumed failure of the auxiliary feedwater to initiate was analyzed. This break size is equal to three stuck-open PORVs and represents the largest isolable pressurizer break in a W-designed plant.

From the transient calculation, minimum inventory was reached at 4800 seconds (80 minutes), at which time, system pressure was approximately 900 psia, and vessel mixture level was just over two feet above the core. PORV isolation at this point in the transient results in a slight vessel level reduction to 14 feet, or two feet above the core, followed by a rapid mixture level rise, due to the termination of inventory loss through the PORVs. The subsequent repressurization results in the eventual re-establishment of natural circulation heat removal by 7800 seconds (130 minutes) to the remaining water in the secondary sides of the steam generators, demonstrating that an interval of an hour or more would be available for operator actuation of AFW following PORV isolation for such an accident.

In this case AFW activation at the time of PORV isolation, would result in a continuing depressurization, with natural circulation established at 7000 seconds, and accumulator activation at 8000 seconds. For the two cases summarized above, and the case in which no operator action was taken, no uncovering of the core was predicted.

A letdown line break represents a two-inch cold leg break. The time to minimum inventory occurs at 1900 seconds (32 mins.), with system pressure near 1100 psia and core mixture level at just over five feet. Letdown line isolation at this point results in rapid recovering of the core and continuing refilling of the vessel to the hot leg elevation by 3000 seconds (50 minutes) into the transient.

From the results of these analyses, it is concluded that no adverse uncovering of the core would occur as a result of operator isolation of the small breaks, even if such isolation should occur at the most inopportune time in terms of system inventory.

4.2.2.3.7 Equipment Used in Small Break LOCA Analyses

The W small break LOCA analyses discussed above assumed the operation of various equipment which has not been previously characterized as part of the reactor protection system or part of the engineered safety features. The equipment used to trip the reactor coolant pumps, the pressurizer relief valves, and equipment used to automatically actuate the pressurizer relief valves, and equipment used to remotely control the pressurizer relief and block valves fall into this category. The reliability and redundancy of these systems should be reviewed and upgraded, if needed, to provide appropriate protection. These systems should also be qualified for the post-LOCA environment consistent with Recommendation 9 of NUREG-0585.(5)

4.2.3 Plant Simulators

The small break LOCA analyses performed by W indicate that several modes of primary system depressurization are possible. It is important that training programs expose the reactor operators to the various kinds of system transient behavior and symptoms of inadequate core cooling. However, in previous simulator training, only one small break LOCA was included in the program. For this reason, it is recommended that plant simulation for operator training offer, as a minimum, the following small break LOCA events:

(1) continuous depressurization,
(2) pressure stabilized at a value close to secondary pressure,
(3) repressurization,
(4) stuck open pressurizer relief valves(s), and
(5) stuck open letdown valve.

Each of these cases should be simulated with reactor coolant pumps running as well as with the pumps turned off. The first three events should be simulated for both cold and hot leg break locations. In addition to assuming single failures in the ECCS and feedwater systems, extended (main and auxiliary) loss of feedwater should also be simulated in conjunction with the above events.

4.2.4 Staff Audit Calculations

4.2.4.1 Introduction

The methodology employed by W for small break evaluations is described in WCAP-8200, Revision 2 (23). The computer program developed for this purpose is designated WFLASH. The adequacy of the model has been discussed further in Section 2.0 of WCAP-9600,(3) in response to our concerns on small breaks resulting from the TMI-2 accident. Increased attention has been focused on small break LOCA behavior and overall system response to such breaks.

The primary purpose of the audit comparison is to provide reasonable assurance that the calculated system response obtained using the WFLASH computer code can be used as a base for guidelines in the development of operator training and plant emergency procedures to be used to detect and to mitigate the consequences of a small break LOCA.

Three types of system transients are studied. They are a depressurization, a pressure hang-up, and a re-pressurization transient. The corresponding break sizes are a four-inch diameter (0.087 ft.^{2}), a one-inch diameter (0.005 ft.^{2}), and a 1/2-inch diameter (0.0014 ft.^{2}) break. The break location is assumed to be in the pump discharge leg.

The model and assumptions used in our audit analyses are described in Section 4.2.3.2 below. The differences between the RELAP4 and the WFLASH analyses, which have a significant effect on the analyses, are discussed in Section 4.2.3.3, and the comparison of the analyses for the four inch diameter break are presented in Section 4.2.3.4.

4.2.4.2 NRC Staff Audit Analysis Model and Assumptions

Our audit analyses were performed with RELAP4/MOD7.(33) The following analytical options were used:

(1) Compressible flow with momentum flux is used at all junctions, with the exception of those junctions between the reactor vessel and the hot and cold legs, pressurizer and accumulator junctions, core bypass junctions, and all fill junctions. These junctions use incompressible flow with no momentum flux.

(2) Vertical slip is used in all vertical junctions in the model, except in the steam generator U-tubes.

(3) The Wilson bubble rise model is used in all reactor vessel volumes (with the exception of the core bypass regions), the pressurizer, and the pump suction volumes. A bubble gradient of 0.8 is used. Complete phase separation is modeled in the accumulators. A constant bubble rise velocity and bubble gradient are calculated to achieve an initial energy balance for use in the steam generator secondaries.

(4) The pressurizer surge line is lumped into the pressurizer volume. Single volume pressurizers are currently used for all cold leg break analyses.

(5) The Henry-Fauske/Moody critical flow option is used. A multiplier of 1.0 is applied to both the subcooled and saturated flow regimes.

(6) The new slip model developed for RELAP4/MOD7 is utilized. The new model employs a flow regime-dependent correlation, which results in a more accurate value for interphase slip velocities.

(7) The RELAP4/MOD7 self-initialization routine is used to obtain an initial system pressure and energy balance.

(8) The non-equilibrium model developed for RELAP4/MOD7 is not used until emergency core cooling flow is initiated. However, if deleterious non-homogeneous condensation effects occur prior to emergency core cooling flow initiation, the model could be used. The model allows coexistence of subcooled emergency core cooling water with primary system steam in a single volume.

(9) Decay heat is calculated using 1.2 times the American Nuclear Society ANS (5.1) Standard. The RELAP4 kinetics package is used to determine reactivity effects due to fuel and moderator density feedback, and to calculate scram reactivity based on an explicit time versus reactivity insertion table.

(10) The steam generator secondaries utilize the natural convection option for heat transfer.

(11) Reactor trip and primary coolant pump trip are modeled to occur on low primary system pressure (1825 psia plus a 3.4 second delay).

(12) The break opening is located at the cold leg centerline elevation.

The nodalization diagram for the cold leg break analyses is shown in Figure VIII-3.

4.2.4.3 Model and Modeling Differences

The analyses performed by W were for a RESAR PWR system. Our model utilized data from the BE/EM studies[23]. The BE/EM model was developed for a four-loop plant, and is geometrically similar to the RESAR configuration.

The BE/EM data was modified, as needed, to be consistent with the W analyses. Core power, trip set-points, and safety systems were modified. However, in our initial analyses, the primary system coolant flow rate and the secondary side pressure were not modified. The primary system coolant flow rate was 10% higher than the W cases, and the secondary side pressure was 240 psi lower. As a result of these differences, the comparisons of the analyses were not in reasonable agreement.

The effects on the transient response were seen in the behavior of the steam generator secondary side, specifically in the operation of the safety valves. After reactor trip, and subsequent trip of main feedwater, the secondary side pressure increase from the initially too-low value was not great enough to reach the safety valve setpoint. For smaller breaks (less than four inches in diameter), the secondary side pressure will control the primary system response, tending to hangup near the safety valve pressure setpoint.

Another modeling difference which had a significant effect on the calculated transients was the assumption used for high and low pressure safety injection delivery. The total flow delivered was based on the single failure criterion, assuming complete spillage from one of the four loops. However, in our initial calculations, it was assumed that 1/4 of the remaining total flow entered each of the four cold legs. In the W model, 1/3 of the remaining total flow entered each of the three cold legs. The important effect here is the calculation of the break flow. With the cold safety injection water in our analyses, the break flow quality did not exceed about 80%. In the W analyses, the quality reached 100% - a steam release. The critical flow is a function of the quality, and for small breaks, the calculation of the mass and energy released through the break has a very strong impact on the system response.

The non-equilibrium model currently available in RELAP is still in the developmental stages, and has not yet been verified against any integral test data. EG&G believes that there is a problem with the condensation model, and has recommended that further investigations in this area be performed. The predominant effect was observed for the 4-inch diameter case, when the safety injection tanks were actuated and the injection point was the cold leg volume. The current condensation model reduced the primary pressure approximately 360 psi by effectively condensing all the steam in the cold leg volume. This magnitude is inconsistent with both the W calculation and with experimentally observed data (Semiscale test S-02-6). In an attempt to lessen the impact of safety injection tank injection, and obtain a more realistic calculation, either the RELAP 4 non-equilibrium model would have to be changed, or the modeled injection location could be changed (i.e., from the cold leg volume to the downcomer volume, which would be consistent with the W methodology).

Sensitivity analyses were performed with the same model with safety injection into the upper downcomer node and into the lower downcomer node, as shown in Figure VIII-4 for system pressure and in Figure VIII-5 for accumulator injection. Injection into the upper downcomer was selected as being most representative of the experimental observations (as seen in S-02-6). The effects, though lessened, may still be too large. The amount of condensation is directly proportional to the amount of steam in the injection node. No studies using the condensation coefficient as the parametric variable were performed.

The reduction in the primary system pressure determines the rate, and amount, of accumulator water injected, as shown in figure VIII-4b. The reflooding of the core is dependent on this flow. The sensitivity analyses performed demonstrate the influence on accumulator injection. The amount of steam present at the injection location is the predominant factor which determines the accumulator mass delivery. The results of an analysis will be influenced by the model and the modeling used to calculate the accumulator flow. Additional studies would be required to obtain the necessary information to perform an Appendix K to 10 CFR Part 50 analysis. More work in this area is underway at EG&G Idaho, since more recent experimental data, including LOFT Test L3-1, indicate less depressurization than the Semiscale S-02-6 tests.

The differences between the two models, as described above, were corrected and a follow-up analysis for the 4-inch cold leg break was performed. A comparison of the W and staff calculations for the 4-inch break is presented in Section 4.2.3.4 below.

One additional difference was observed during the comparison of the W analyses with our analyses-an energy imbalance. The difference was found to be the result of an input parameter in the WFLASH computer program. The volumetric heat capacity of the fuel is an input in the WFLASH computer program. For a steady-state condition, the energy input of the core should match the energy removal in the steam generators. The values input for these studies were conservatively calculated and resulted in an energy imbalance, i.e., more heat being added to the system than removed. The effect of this imbalance corresponded to a 10 psi per 100 second increase in the secondary side pressure, and was most noticeable in the 1/2-inch break analysis. To correct this situation, W will use more care in obtaining the input data, and for future analyses, W has stated that the data will be verified and a steady-state condition will exist prior to the initiation of a transient.

4.2.4.4 Audit Analyses Comparison

The four-inch diameter cold leg break, using the follow-up staff calculational results, is presented here and compared to the W analysis. The difference between the RELAP4 and WFLASH models, which have a significant influence on the calculated transient, are discussed.

While the intent was to model the same system as W for the audit analyses, the upper head configuration of the RESAR plant is different from the BE/EM plant. The difference is characterized in the flow path from the upper head to the upper plenum region. In the RESAR model, the fluid in the upper head does not fall back into the upper plenum (and into the active core region) as rapidly as in the BE/EM model. Therefore, less fluid is available to remove the heat from the core. As will be seen below, the W analyses tend to predict uncovering of the core faster and to a greater depth than the staff analysis, for the four-inch diameter break case. For smaller breaks, the effect of the upper head is seen as a delay in the depressurization characteristics of the analyses. We will perform an analysis with a modified upper head to verify this difference in the two models.

The most significant difference between the two analyses is the treatment of the heat transfer from the reactor core to the primary coolant. When the mixture level falls below the top of the active core, the heat transfer assumption tends to dominate the calculated transient.

The assumption in RELAP4 of thermal equilibrium within the control volume keeps the steam above the mixture level at saturation. While superheating of the steam could occur in a PWR, RELAP4 does not allow superheating within a given control volume (until all the liquid has flashed) but flashes more liquid to saturated steam. A hand calculation indicates that the top region of the core could be superheated by about 360°F. If the model of the core were sufficiently detailed, this superheat would represent an upper bound for the expected increase in the calculated peak clad temperature.

WFLASH also assumes thermal equilibrium within a control volume. However, when the mixture level falls below the top of the active core, WFLASH attempts to treat the heat transfer more realistically by removing the excess energy from the control volume, instead of flashing more liquid to sautrated steam. As a result of removing this energy from the core volume and transferring it to the primary coolant loops and the vessel upper heat region, the system responds in a different manner than if the energy was assumed to be added to the control volume.

When the core begins uncovering, at approximately 400 seconds in the W analysis, the direct heat additions to the primary loops drive the break fluid quality rapidly to 1.0. In our analysis, it can be seen that the heat transfer assumption results in a longer delay in obtaining a break quality of 1.0. Figure VIII-6 compares the break flow calculation for both analyses. Figure VIII-7 compares the reactor coolant system pressure calculated by both models. The re-pressurization observed for the W analysis (and the influence on the break flow) for the period from 200 to 400 seconds is attributed to the flashing of the upper head region, which acts like a pressurizer.

The mixture level within the active core region is shown in Figure VIII-8. Figure VIII-9 compares the clad temperature calculation for both analyses. Note that our results were obtained for the average core, while the W results are for the hot rod.

A preliminary scoping analysis was performed for the hot rod using the HEATO computer code(34). This analysis is a combined manual and computer program method. Currently radiation heat transfer to steam and metal to steam water reactor heat generation are not included. Basically the assumptions used are: (1) all heat generation below the mixture level generates saturated vapor, (2) fifty percent of the heat generation above the mixture level superheats the vapor, and (3) the heat transfer coefficient is based on the fluid temperature. A peak cladding temperature of approximately 1500°F is estimated for this analysis.

Figures VIII-10 and VIII-11 compare the steam generator secondary side pressure and mixture level, respectively. Note that the initial level of 100 feet for the WFLASH analysis is an artifact of the modeling used by W. This value can be converted to the true level, based on the actual geometry of the steam generator.

At approximately 1175 seconds, the accumulators are actuated in the staff analysis. At this time, the large safety injection flow rate turns the heat-up around and refloods the active core. In the W analysis, the accumulators are actuated at about 750 seconds, reflooding the active core.

In general, the audit analyses demonstrated that the depressurization trend and system parameters available to the operator are similar in both analyses. With the exception of the heat-up calculation and core uncovering, our analysis is in good agreement with the W analysis. The influence of superheat and of the accumulator effects should be investigated in more detail to provide a better estimate of the cladding temperature response and core uncovering with the RELAP4 computer program.

Comparisons of the staff's and W's analyses for the one-inch and the 1/2-inch diameter breaks are provided in Figures VIII-12 and VIII-13 for the reactor coolant system pressure. The models we used for our analyses do not include the changes described in Section 4.2.2.3, above.

The audit analyses, again, demonstrate that the pressure hang-up and the repressurization trends are similar for both analyses.

The audit comparisons provide reasonable assurance that the calculated system response obtained from the WFLASH program can be used as a basis for guidelines in the development of the operator training and plant emergency procedures to detect and to mitigate the consequences of a small break LOCA.

4.2.4.5 Conclusions

The following conclusions are based on our audit of the analyses performed by W:

1. The calculated system response to the three break sizes analyzed demonstrates the ability of the computer program WFLASH to predict the expected behavior of a depressurization, a pressure hang-up, and a re-pressurization transient. Reasonable assurance is therefore provided that the calculated system response using WFLASH may be used as a basis for guidelines in the development of operator training and plant emergency procedures to be used to detect and to mitigate the consequences of a small break LOCA.

2. Significant differences in both the methods and the plant designs used by the staff and by W explain most of the differences between the calculations. In our judgement, better agreement in the system transient responses would be obtained if the two plant designs were the same, and if the our model of the core were more detailed to account for the effects of superheat. The above-discussed core uncovering and subsequent heatup calculations performed with the RELAP4 computer program are unrealistic. A non-conservative evaluation of the fuel cladding temperature response could result.

   The treatment of steam superheat and steam generation rates in RELAP4 could result in a peak cladding temperature several hundred degrees too low. Appropriate changes to the program and/or to the modeling procedures employed would have to be made to eliminate the current uncertainties in the core uncovering and heatup calculations.

3. The effects of accumulator injection on the transient should be further investigated to determine the amount of condensation realistically expected and to determine the effect on core recovering and heatup. The condensation model and the modeling procedures (i.e., the injection location used in the computer analyses) require further investigation to assure that the effects of accumulator injection are not biased in a non-conservative manner. Semiscale and LOFT test data should be used to verify the models.

4.2.5 Two-Phase Natural Circulation and Accommodation of Loss-of-All Feedwater

As a result of our review of small break analyses, we have concluded that for small breaks which cannot remove all of the decay heat, repressurization of the primary system will occur without secondary system heat removal. For plants with low shutoff head HPI pumps, repressurization above their shutoff head would make them ineffective to supply make water to the primary system. Unless action is taken to reduce the primary system pressure to below the HPI pump shutoff head, core uncovering and fuel damage could result.

The need to provide additional relieving capacity for the primary system has also been identified as part of the anticipated transients without scram (ATWS) study, and more recently as part of the staff's post-TMI-2 requirements to provide venting capability at the primary system high points.

Loss of secondary heat removal could occur in two ways: loss of all feedwater, and failure to establish or maintain natural circulation.

To date, the PWR industry has not provided any data to experimentally verify its analytical predictions of two-phase natural circulation. Therefore, we will require that verification of the various modes of two-phase natural circulation, and the transitioning between modes predicted by vendor analytical models be verified against appropriate experimental data.

Notwithstanding the ability of the PWR industry to verify its prediction of two-phase natural circulation, the staff believes that a diverse heat removal path, independent of the secondary system is desirable. To this effect the NRC is considering in its TMI-2 Action Plan this and other system changes designed to improve overall plant safety and reliability. Specific Commission policy and schedule will be established subsequent to Commission adoption of this plan.

Conclusions:

(a) We find that the predicted flow through the PORVs has a large uncertainty when the flow is two-phase in composition.

Because of this high uncertainty, we cannot conclude that for plants with low shutoff head HPI pumps, the primary system can be depressurized to initiate HPI flow in sufficient time to preclude unacceptable core uncovering should secondary heat removal capability be lost. Section 2.1.2 of reference 1 requires that both relief and safety valves be qualified under conditions of both subcooled and two-phase flow.

(b) A diverse decay heat removal path, independent of the steam generators is desirable. This desirability stems from the fact that although the probability of losing either all feedwater or natural circulation is considered low, it is nevertheless unquantified and finite.

Recommendations:

(a) The NRC TMI-2 Action Plan should consider the need for a diverse decay heat removal path independent of the steam generators. Consideration of diverse systems should include, for example, (a) increased PORV relieving capacity (b) higher shutoff head HPI pumps, or (c) installation of a high pressure residual heat removal system.

(b) If a system which manually depressurizes the primary system to below the HPI actuation pressure is selected, then the time available to the operator to decide if system depressurization is necessary (i.e., feedwater cannot be restored) should be greater than 20 minutes. We believe that times less than 20 minutes do not provide the operator sufficient time in which to full analyze the situation, and could result in incorrect action being taken.

FIGURE VIII-3

RELAP4/MOD7 Nodal Diagram Westinghouse 4-Loop PWR
Small Break Model

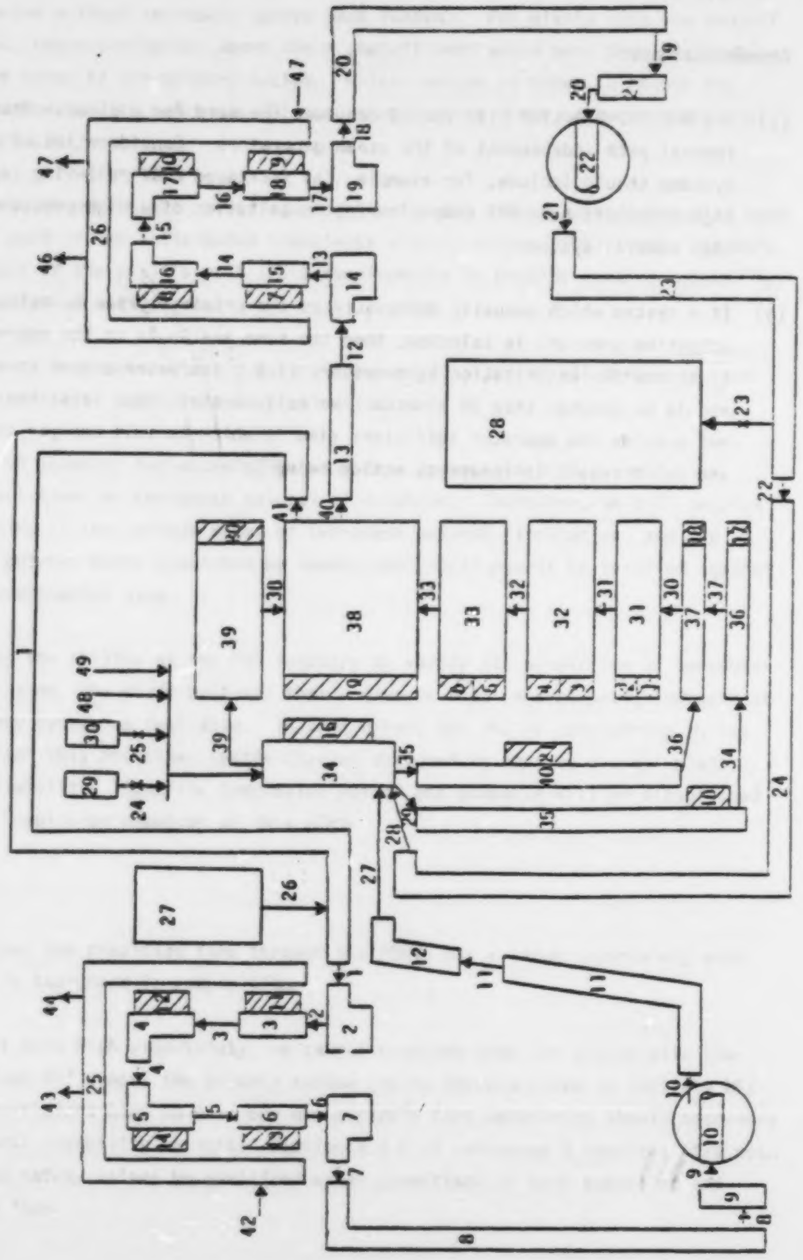

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Figure VIII-4

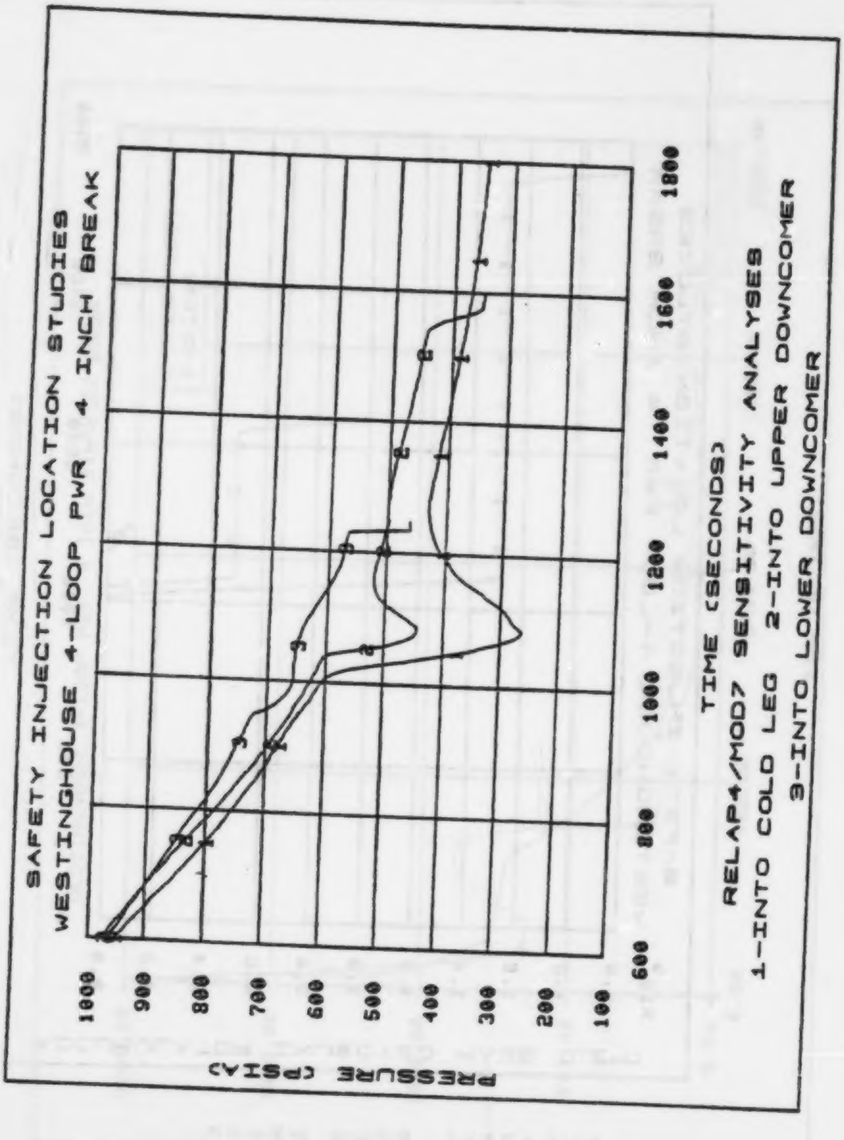


Figure VIII-5

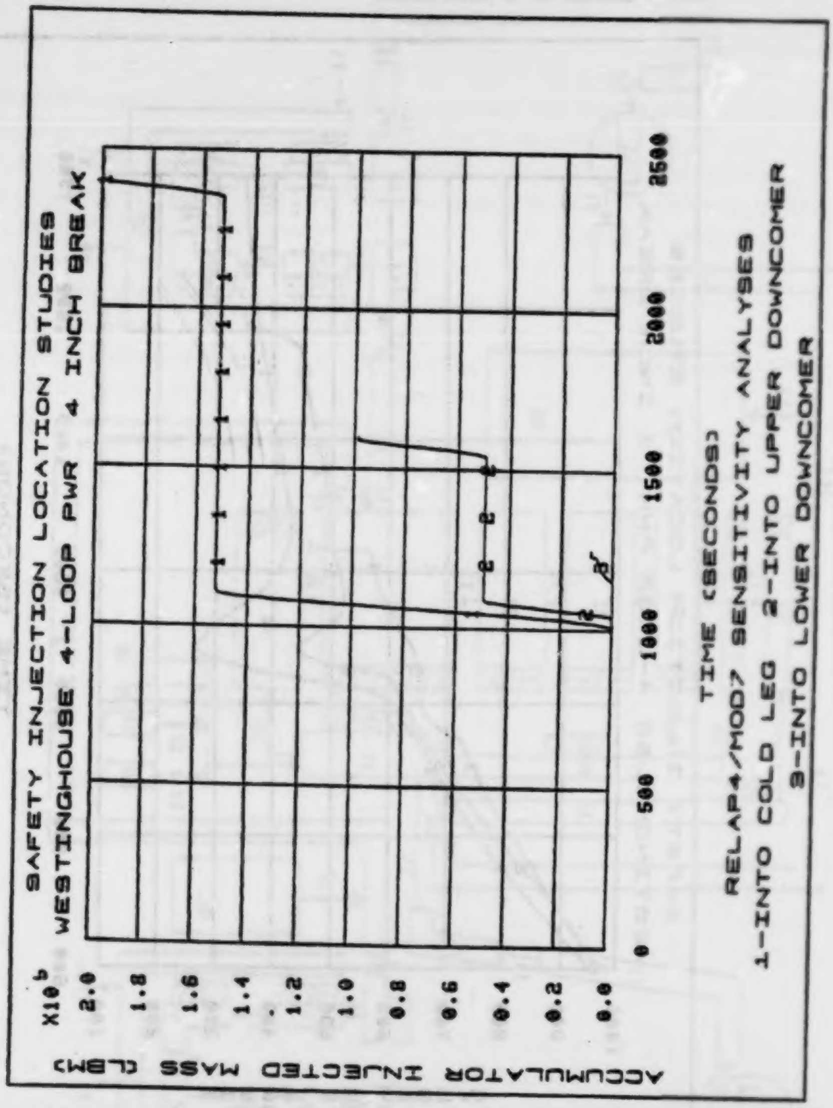


FIGURE VIII-6

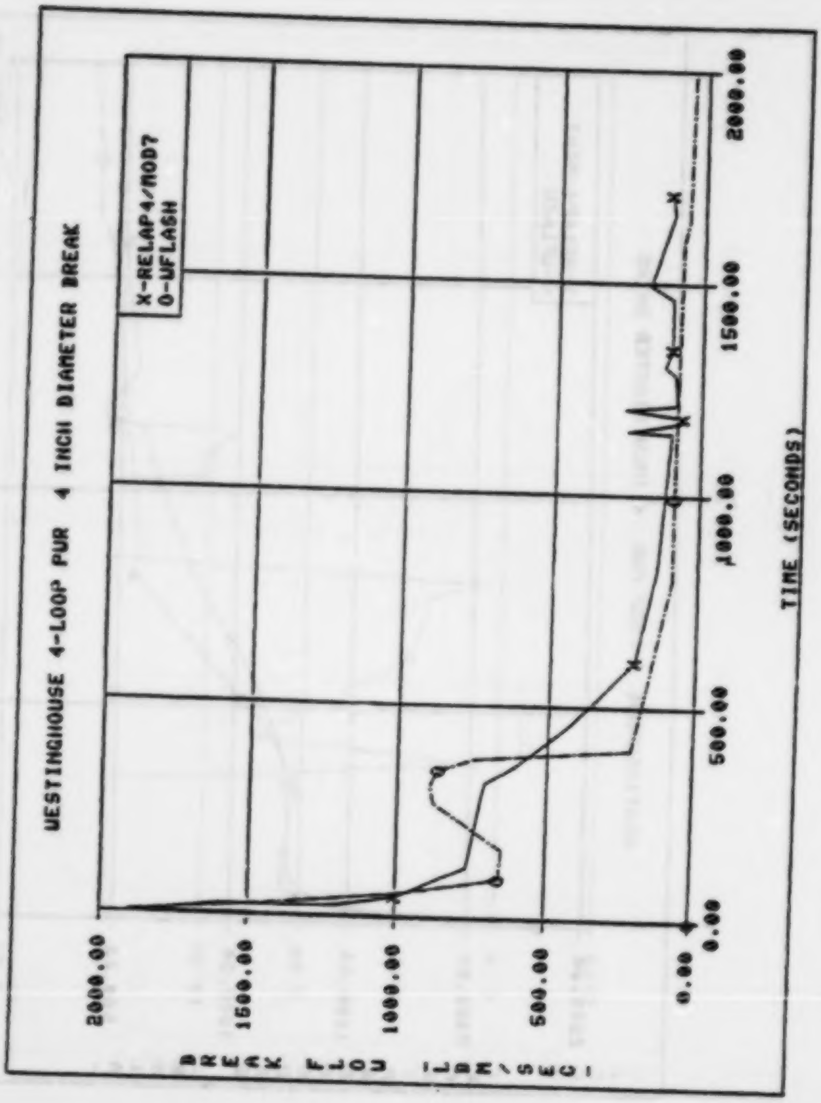


FIGURE VIII-7

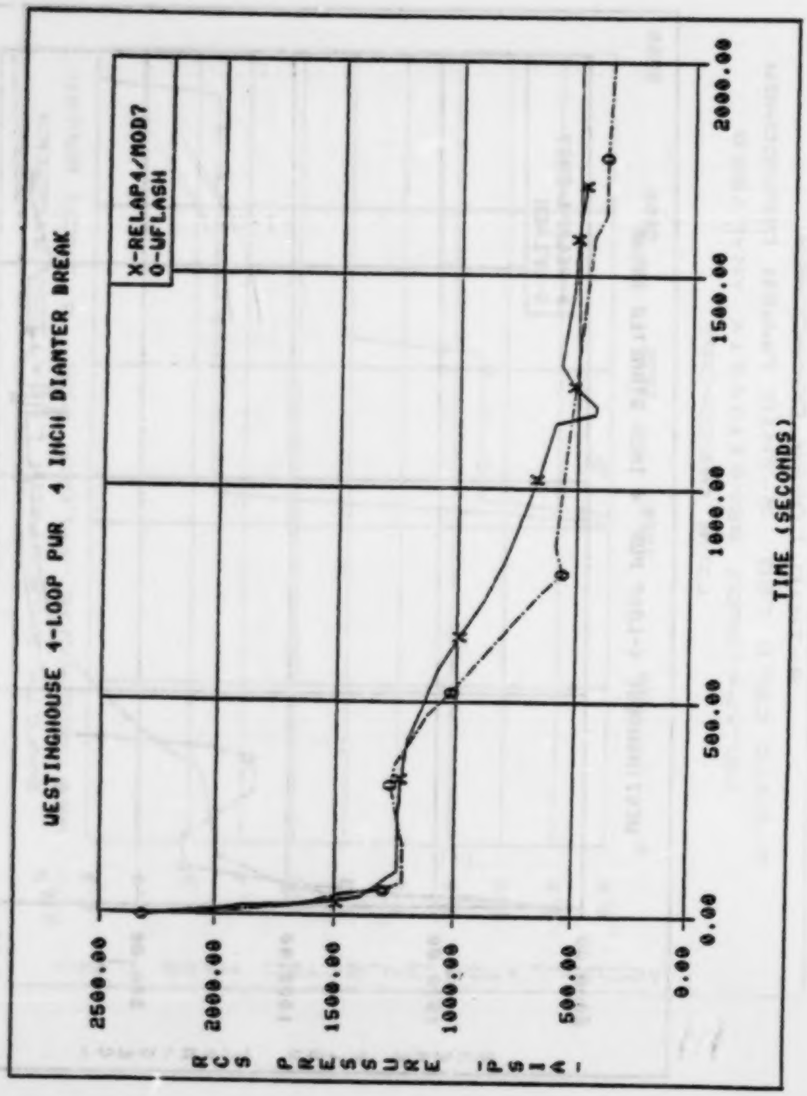


FIGURE VIII-8

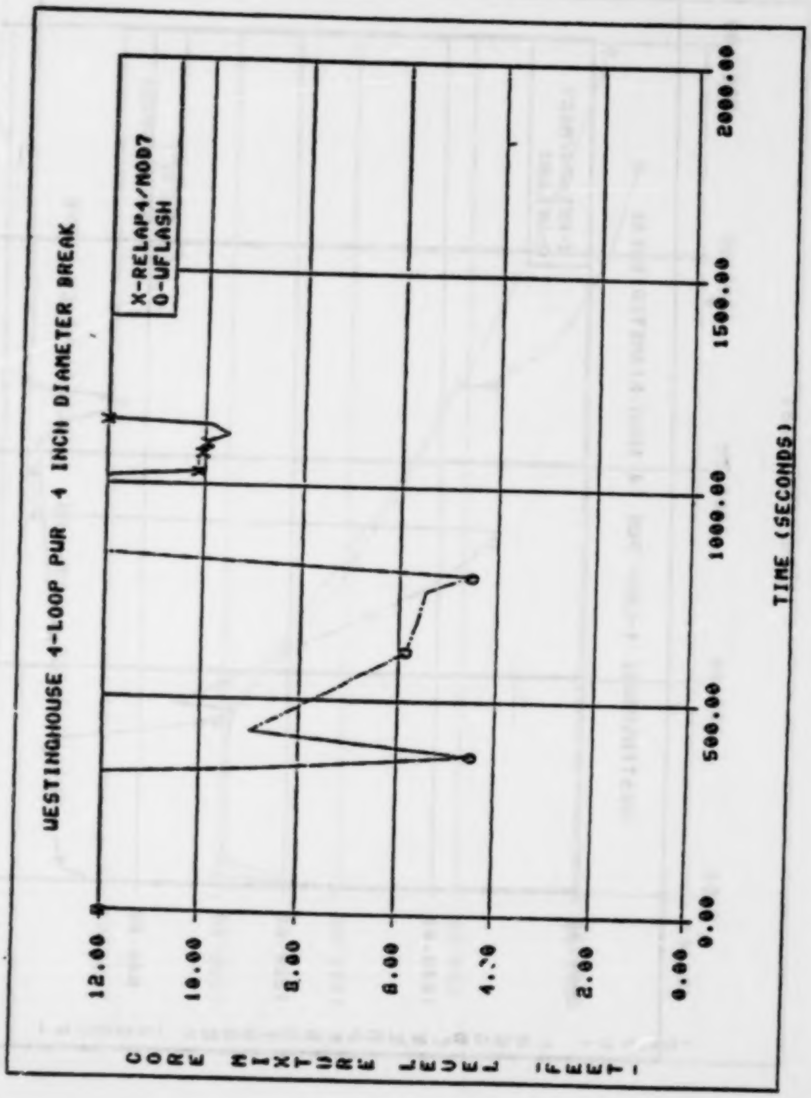


FIGURE VIII-9

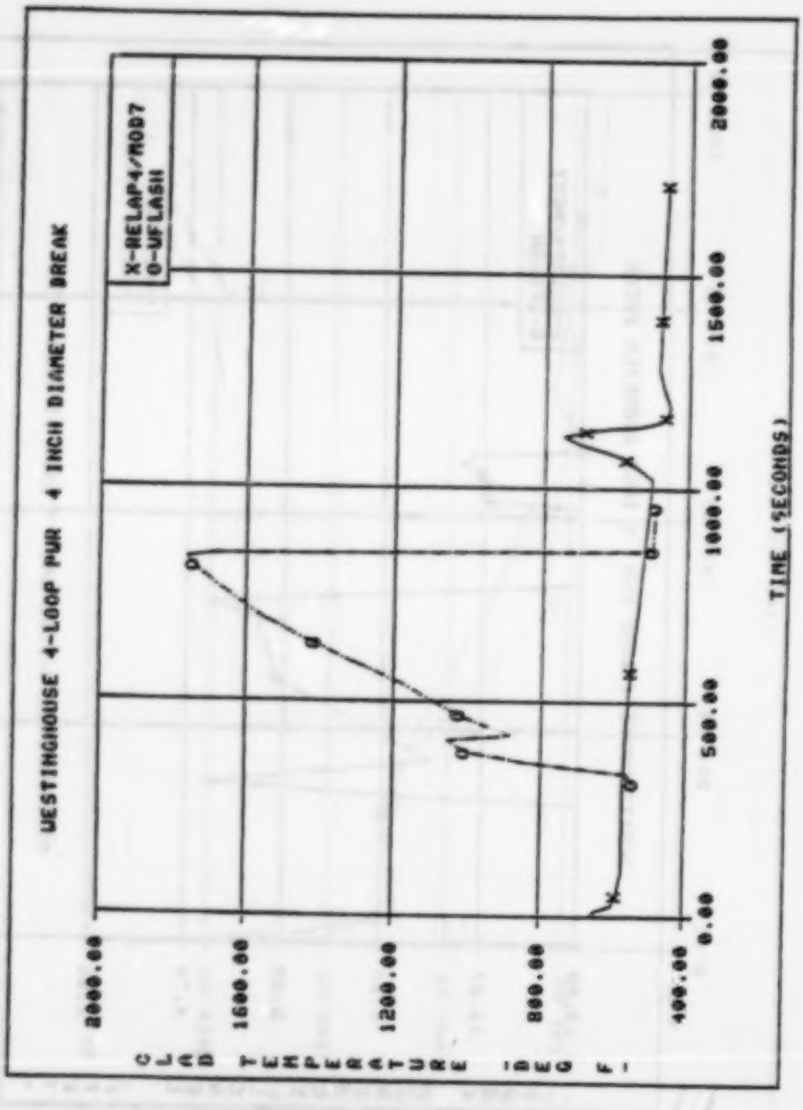


FIGURE VIII-10

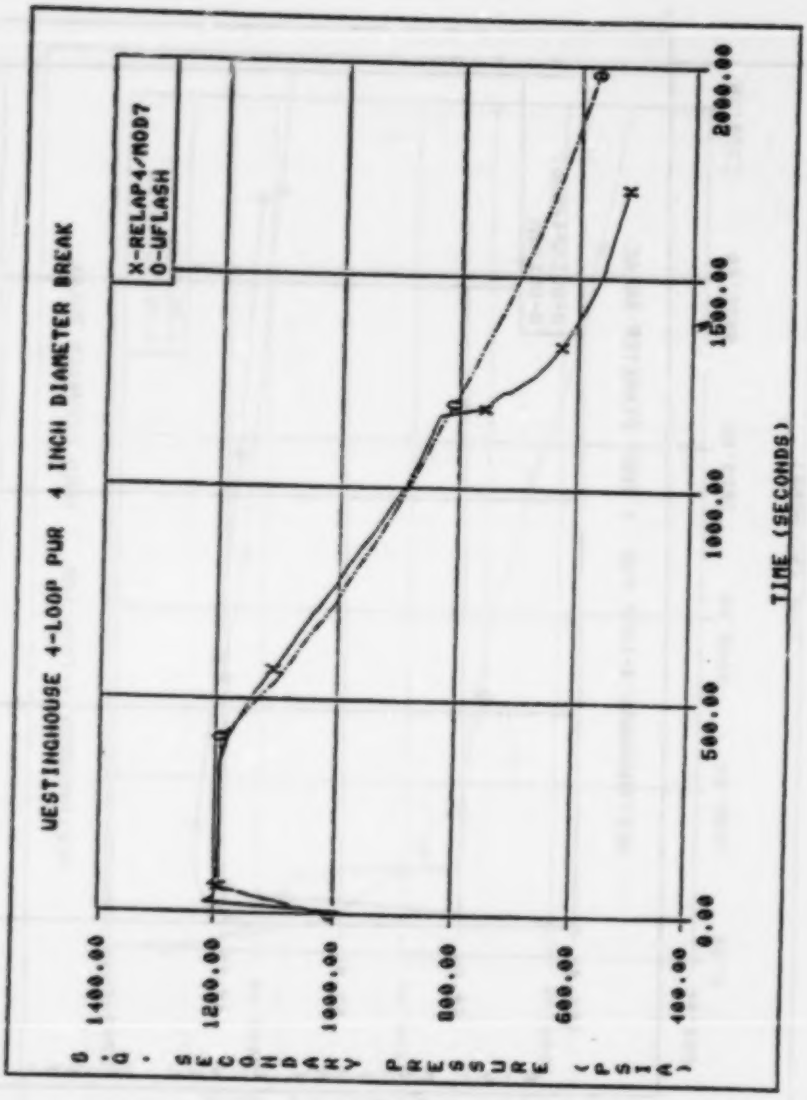


FIGURE VIII-11

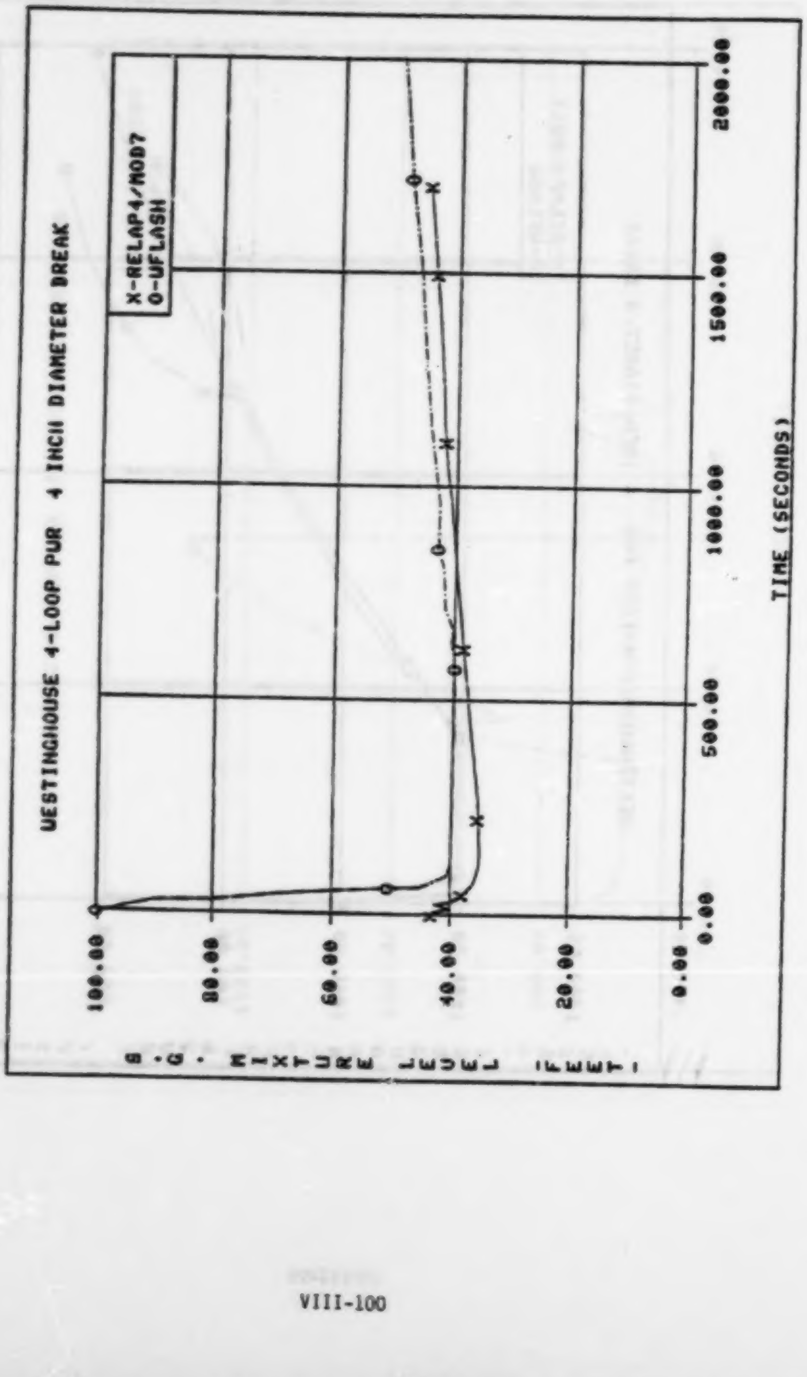


FIGURE VIII-12

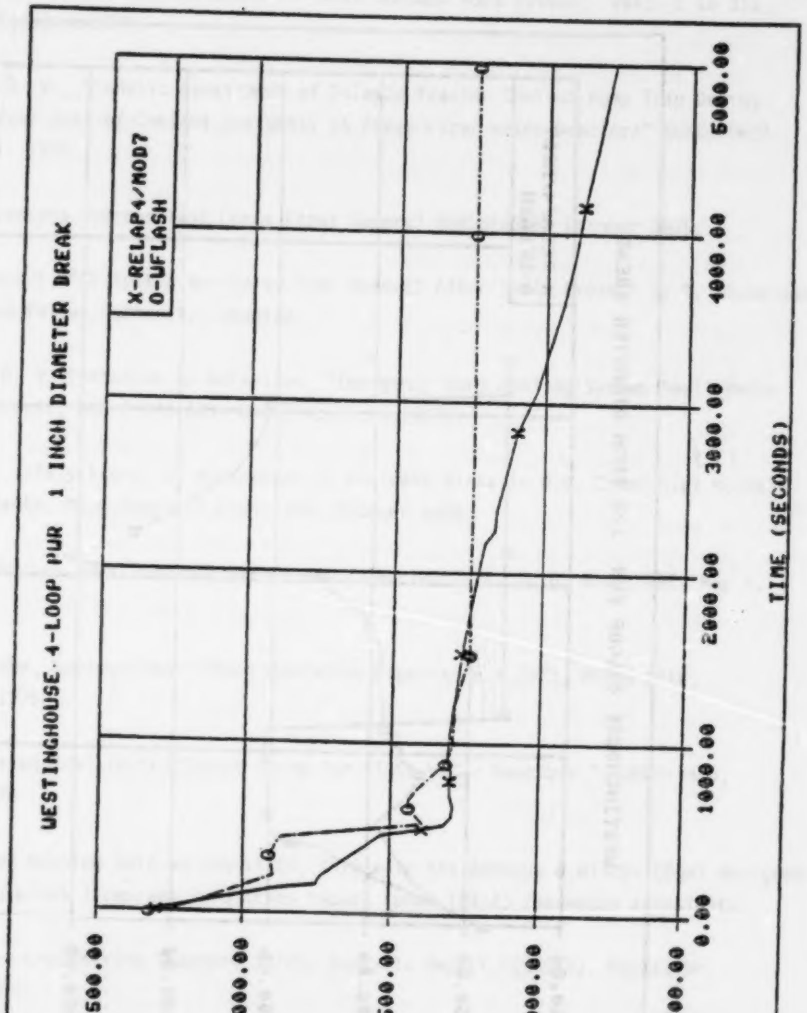
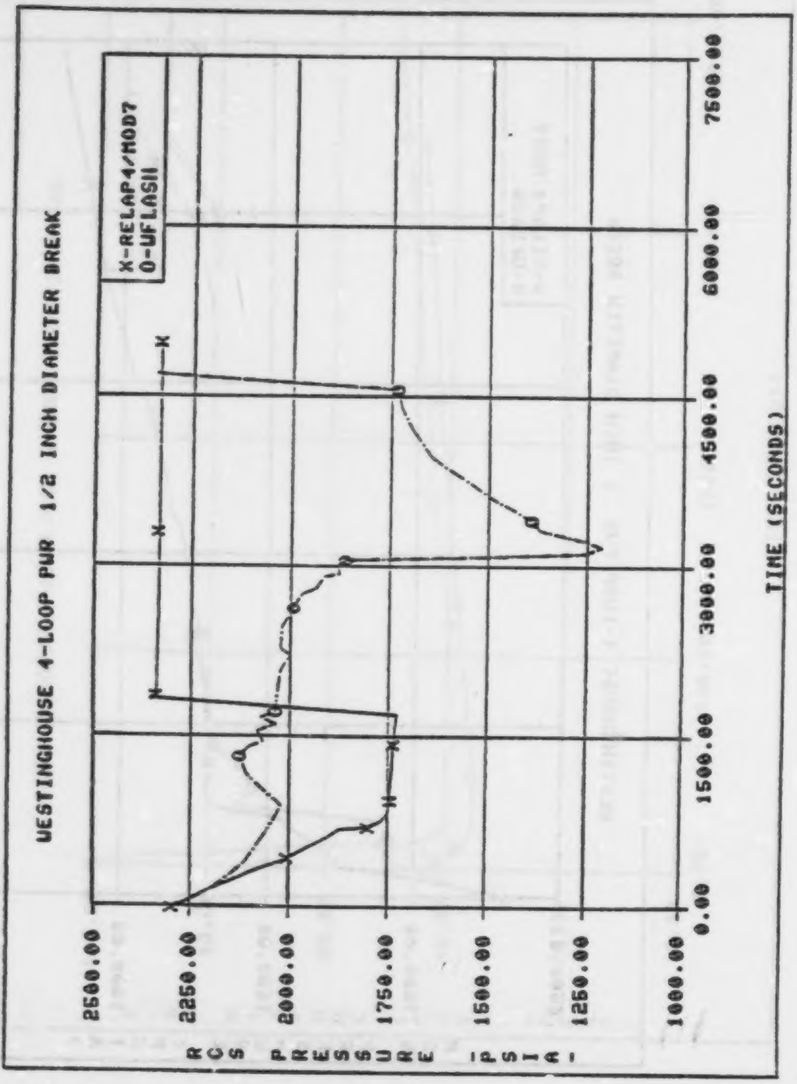


FIGURE VIII-13



5. REFERENCES

1. TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations. NUREG-0578, July 1979.

2. Letter, D. F. Ross, NRC, to T. M. Anderson, Westinghouse, dated June 4, 1979.

3. "Report on Small Break Accidents for Westinghouse NSSS System," Vols. I to III, WCAP-9600, June 1979.

4. Sheron, B. W., "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break loss-of-Coolant Accidents in Pressurizer Water Reactors" NUREG-0623, November, 1979.

5. "TMI-2 Lessons Learned Task Force Final Report" NUREG-0585, October 1979.

6. Draft report, "CE System 80--Decay Heat Removal After Small Break," by C. Michelson, Tennessee Valley Authority, undated.

7. Letter, D. R. Patterson to McFarland, "Emergency Core Cooling System Small Break LOCA Analysis, N4M-2-149(AR), dated April 26, 1978.

8. "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, Main Report," WASH-1400, October 1965.

9. T. M. Anderson, Westinghouse letter number NS-TMI-2078, to D. Ross, NRC, May 1, 1979.

10. M. R. Beebe, Nuclear Power Plant Operating Experience - 1978, NUREG-0618, October 1979.

11. "Anticipated Transients Without Scram for Light Water Reactors," NUREG-0460, April 1978.

12. Summary of meeting held on August 28, 1979 with the Babcock & Wilcox (B&W) designed operating plant licensees to discuss recent (post TMI-2) feedwater transients.

13. Combustion Engineering Standard Safety Analysis Report (CESSAR), Docket No. STN. 50-470.

14. Letter: D. F. Ross, Jr., B&OTF, to C. Reed, dated November 5, 1979.

15. Letter, D. A. Ross, Jr., NRC, to Cordell Reed, W Owners Group, dated December 6, 1979.

16. Letter, D. A. Ross, Jr., NRC, to Cordell Reed, W Owners Group dated December 27, 1979.

17. G. P. Lilly, A. G. Stevens, and L.E. Hochreiter, "Mixing of Emergency Core Cooling Water With Steam: 1/14-Scale Testing Phase," EPRI 294-2, January 1975.

18. G. P. Lilly, and L.E. Hochreiter, "Mixing of Emergency Core Cooling Water with Steam: 1/3-Scale Test and Summary," EPRI 294-2, June 1975.

19. Letter, L. P. Leach to R. E. Tiller, "Pressure Oscillations Resulting from ECC Injection," LPL-47-77, dated April 20, 1977.

20. "Task Action Plans for Generic Activities - Category A," NUREG-0371, November 1978.

21. Memorandum, C. C. Graves, DSS to D. F. Ross, Jr. B&OTF, "Pressurizer Power Operated Relief Valves," November 1, 1979.

22. Rohsenow, W. M., and Hartnett, J. P., "Handbook of Heat Transfer" McGraw-Hill, 1973, Pg. 12-9.

23. V. J. Esposito, K. Kesavan, and B. Maul, "WFLASH - A Fortran IV Computer Program for Simulation of Transients in a Multi-Loop PWR," WCAP 8200, Rev. 2, June 1974.

24. Delaney, H. M., "United States Standard Problem 6 and International Standard Problem 4 Final Report," CVAP-TR-4-78, dated March 1978.

25. Powell, A. W., "Flow of Subcooled Water Through Nozzles," WAPD-PT(V), April 1961.

26. Sozzi, G. L., and Sutherland, W. A., "Critical Flow of Saturated and Subcooled Water at High Pressure, "NEDO-13418, July 1975.

27. Simon, U., "Blowdown Flow Rates of Initially Saturated Water, CONF-730304, PP. 172 -195, March 1973.

28. Ericson, L., The Marviken Full-Scale Critical Flow Test Interim Reports, Results from Test 1-16, March 1978 - May 1979.

29. Slater, C. E., "Comparison of Predictions from the Reactor Primary System Decompression Code (RELAP-3) with Decompression Data from the Semiscale Blowdown and Emergency Core Cooling CECC) Project, USAEC Report, TN-144, 1970.

30. Cunningham, J. P. and Yeh, H. C., "Experiments and Void Correlation for PWR Small Break LOCA Conditions," Trans. ANS, Volume 17, 369 (1973).

31. Skwarek, R., Johnson, W., Myer, P., WCAP-8470-RA, "Westinghouse Emergency Core Cooling System Small Break, October 1975 Model," January 1979.

32. "Analysis of Delayed Reactor Coolant Pump Trip During Small Loss-of-Coolant Accident for Westinghouse Nuclear Steam Supply Systems," WCAP-9584, August 30, 1974.

33. G. N. Johnson, et. al., "RELAP4/MOD7 (Version 2) User's Manual", CDAP-TR-78-036, EG&G, Idaho, August 1978.

34. "Guide to the HEATO Code", Semiscale Program, SEMI-TR-003 EG&G, Idaho Falls, Idaho, March, 1979.

35. Moody, F. J., "Maximum Flow Rate of a Single Component Two Phase Mixture, "Journal of Heat Transfer, February 1965.

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APPENDIX IX

OPERATOR TRAINING AND OPERATING PROCEDURE CONSIDERATIONS

1. TRAINING

1.1 Finding (previously stated in NUREG-0560)

Operator training has evolved over the last ten to fifteen years from concentrated on-the-job training programs, with little time allotted to formal training, to the current more formal, NRC-approved programs. In addition, the expanded use of simulators has contributed significantly to the quality of operator training.

The staff's Operator Licensing Branch (OLB) has used simulators in its examination of applicants for operator licenses for approximately five years. The OLB finds that such examinations are much more demanding than normal "walk-thru" type examinations. The examinations are more demanding on both the applicant and the examiner with respect to having a keen understanding of the nuclear system. As a result, a better evaluation of an individual's operating ability can be made by using a simulator. However, the extent of the improvement in evaluation potential in each case is highly dependent on the degree of similarity between the simulator and the plant that the individual will actually operate.

In the past, training programs have underemphasized nonstandard, passive conditions, such as misaligned systems, undetected failures of engineered safety features (ESF) equipment and multiple failures. Irrespective of the merits of the single failure criterion as a design basis, it should not be considered as a limiting basis for training purposes.

It is generally acknowledged by the staff and the operators themselves that simulator operation is a valuable part of operator training. This consensus is reinforced in EPRI Report No. NP-309, which reads, in part, "Operators regard simulators as the best vehicle for obtaining operational training . . . it helps you to see casualty modes." It is also apparent from the TMI-2 accident that transient recognition by the operator and the operator response based upon his understanding of the plant status are essential to reactor safety. We believe that a primary part of operator training in event recognition and response should be actual "hands-on" operation in response to various plant transients and accidents. This sort of experience can be gained, to some degree, through actual plant operation and walk throughs, but must include event simulation and actual operator response and observation to be most meaningful.

1.2 Recommendations

Based on our review of operator training at operating reactors, we recommend that:

a. All licensed operators be required to participate in a simulator training program to observe such events as a stuck-open power operated relief valve (PORV) and natural circulation. Training on protecting the core should be emphasized on all plants. This includes the means to recognize that an adequate heat sink, primary system inventory, and intact primary and secondary system exist. Simulator training programs should be reviewed to assure that they include the operator errors and equipment failures that contributed to the TMI-2 accident. An evaluation of the simulator control board design and simulated response as compared to the operator's individual response and actual control board design must be made on a case-by-case basis. The differences which may exist must be addressed as part of the operator's training so that negative training feedback will not result.

b. As stated in Appendix VIII, plant simulators should offer, as a minimum, the following small break LOCA scenarios:

(a) Continuous depressurization.
(b) Pressure stabilized at a value close to secondary system pressure.
(c) Repressurization.
(d) Stuck-open PORV.
(e) Stuck-open letdown valve.

Each of these cases should be simulated with the reactor coolant pumps running and with the pumps not running. The first three events listed above should be simulated for breaks in the hot leg and in the cold leg. In addition to the usual single failures assumed in the ECCS and feedwater systems, the extended (main and auxiliary) loss of all feedwater should be considered (see Section 4.2.2.1 of Appendix VIII).

2. OPERATING PROCEDURES

2.1 Background

Operating and emergency procedures are developed in accordance with Regulatory Guide 1.33, Appendix A, "Quality Assurance Program Requirements (Operation)," and Sections 5.3.2 and 5.3.9 of ANSI 18.7 and ANS 3.2, entitled, "Administrative Controls and Quality Assurance of Operation of Nuclear Power Plants."

Each normal operating procedure involves the use of checklists and is based on a controlled evaluation giving final conditions as goals to achieve. On the other hand, abnormal and emergency procedures are completely different, in that the operator is now confronted with automatic responses for which he may have to take manual actions. Therefore, when writing the abnormal and emergency procedures, consideration should be given to the real time that it takes for systems to respond and for the operator to perform a manual function. (The preceding material was previously stated in NUREG-0560.)

The staff requested the plant emergency procedures for loss of coolant, steam line break, loss of offsite power and loss of feedwater events from all operating reactor licensees. Most licensees with W-designed plants complied with this request. A review of emergency procedures for ten W-supplied power reactor facilities (some prior to the TMI-2 accident, some after) indicated deficiencies in providing specific operator guidance to monitor, interpret and respond to critical plant conditions. In general, the procedures failed to guide the operator to monitor and interpret available instrumentation to verify that (1) reactor coolant system inventory is being maintained, (2) the core has adequate flow for heat removal, and (3) a heat sink is available and operating, therefore assuring the capability for heat removal from the reactor coolant system. For example, the emergency procedures for loss-of-coolant accidents (LOCAs) (prior to TMI-2) state that decreasing pressurizer pressure and level are indicative of a LOCA. As emphasized at TMI-2, this is clearly not the case for breaks (for stuck-open valves) in the pressurizer steam space, for which pressurizer level will not be a valid representation of reactor coolant system inventory. IE Bulletins, issued after the TMI event, directed licensees to take action on these and other areas. Licensee responses to these bulletins have been evaluated and separate reports will be issued containing the staff's evaluation. Licensees have in general revised or are revising procedures as a result of these IE Bulletins.

2.2 Guidelines for Emergency Procedures

In response to IE Bulletin 79-06A, a Westinghouse (W) Interdisciplinary Task Force was formed to prepare guidelines for operators for small break loss-of-coolant accidents (LOCAs). The Task Force consisted of safety analysts, systems analysts, training personnel and members of other disciplines. The guidelines which were developed were reviewed and approved by the Working Group on Procedures, which is a subgroup of the Westinghouse Operating Plants Owners' Group.

Preliminary guidelines were submitted to the NRC staff by the Owners' Group as part of the generic report WCAP-9600, "Report on Small Break Accidents for Westinghouse NSSS System." The intent of the guidelines was for each of the utilities using a W-designed nuclear steam supply system to revise or develop its emergency procedures for the operators to use in diagnosing and responding to a loss of reactor coolant. The reference instructions developed by W were expanded to include all emergency events in which the emergency core cooling system (ECCS) was automatically actuated. The guidelines include Immediate Actions and Diagnostics (E-0), Loss of Reactor Coolant (E-1), Loss of Secondary Coolant (E-2), and Steam Generator Tube Rupture (E-3). To date, we have reviewed only E-0 and E-1 for both the 412 Standard Plant, which has high head safety injection pumps, and other plants with nominal 1400 psi safety injection (SI) pumps.

The philosophy of the instructions was for the operator to respond to an event in which safety injection was initiated and, following the required immediate actions, to diagnose the event and perform the necessary subsequent actions. The immediate actions consist of verifying that the automatic actions did occur. Verification, in this context, includes performing the action manually if it dod not occur automatically. These actions are intended to assure that the reactor is adequately shut down, that the safety injection system is performing its design function, and that auxiliary feedwater is being delivered to the steam generators as a heat sink for the core decay heat.

By letter to the W Owners' Group dated November 5, 1979, we approved the generic guidelines E-0 and E-1. The following evaluation has been excerpted from the enclosure to the letter.

Evaluation

The NRC staff reviewed the guidelines with respect to critical operator actions, namely:

1. reactor coolant pump trip.
2. HPI termination criteria.
3. verification of safety systems actuation.
4. verification of a heat sink.
5. monitoring of important system parameters.

During our review, the staff identified modifications to be made to the guidelines to enhance the directions to the operator. These modifications were subsequently incorporated in the guidelines as defined by Revision 1, dated October 16, 1979 and revisions dated October 31, 1979, and November 2, 1979.

The criteria for tripping the reactor coolant pumps are consistent with the analyses presented in WCAP-9584, which have been reviewed by the staff and found acceptable.* In order to implement the criteria in individual plant procedures, each licensee must document the basis for the low pressure set point. This documentation should include defining the steam generator safety valve set points and system and instrument uncertainties associated with the plant. Based on our review of WCAP-9584 and the requirement for each licensee to justify the low pressure trip point described in the preceding section, we conclude that the reactor coolant pump trip criteria are acceptable.

Although we find that the reactor coolant pump trip criteria are acceptable, manual tripping of the pumps should be considered only a short-term solution. For the long-term, we will require that this trip be made automatic.

The criteria for terminating HPI flow are based on a combination of system pressure, subcooling pressurizer level, and steam generator water level. The staff concurs that these criteria are sufficient for establishing subcooled conditions in the core so that HPI can be safely terminated without concern for detrimental voids being formed in the primary system. In implementing these criteria, each licensee is required to document the instrument uncertainties (even in an adverse environment) to show that the criteria in the guidelines will indeed insure subcooled conditions. Based on the above requirement, we find the HPI termination criteria acceptable.

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*See NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors," B. Sheron, November 1979.

As part of the immediate actions, the operator is directed to verify that the ECCS, auxiliary feedwater (AFW), and containment isolation systems have been actuated. We concur that these verifications are sufficient to insure minimum safeguards availability needed to mitigate small break LOCAs.

The operator is also directed to verify that he has established heat removal from the steam generator. We concur that this is a necessary instruction for mitigating small break LOCAs.

The operator is directed to monitor primary system pressure, pressurizer level, and coolant hot leg temperatures to insure that subcooling is maintained, if HPI has been terminated. We concur that monitoring these system variables is sufficient to maintain adequte subcooling in the primary system.

The staff has not reviewed the guidelines for switchover from injection to recirculation or hot leg injection, because these actions are mostly plant-specific instructions. The staff requires each licensee to justify the procedures for switchover to assure that the valve realignments can be accomplished before the RWST is emptied. This justification should include instrument uncertainties and show that the pumps will be protected against operating with inadequate suction. We will require that plants with nominal 1400 psi range SI pumps demonstrate that these pumps will not be deadheaded when in the recirculation phase.

The staff noted that the guidelines are based on obtaining at least minimum safeguards operation to mitigate small break LOCAs. We require each licensee to extend the emergency procedures to cover the loss of all feedwater. Procedures for this degraded condition should also take into account pressure vessel integrity considerations. The W Owners' Group has committed to prepare guidelines for operational proceudes regarding the loss of all feedwater as part of its effort on the issue of inadequate core cooling.

The staff also required that the emergency procedures include instructions for monitoring and initiating (if lost) natural circulation for small break LOCAs where heat removal by the steam generators is required.

The guidelines for such procedures should direct the operator to initiate a controlled plant cooldown is stable system conditions can be maintained. The staff requires that provide procedures for cooling down the plant under natural circulation conditions. These procedures should address boration control and monitoring, cooldown of the pressurizer, and adequate criteria for monitoring coolant system temperatures to insure that voids do not form in the primar system which could inhibit adequate heat removal. As in the case of loss of all feedwater, the W Owners' Group has committed to prepare guidelines for emergency procedures regarding natural circulation and cooldown under natural circulation conditions as part of its effort on inadequate core cooling.

Conclusion

Based on our review, we conclude that the guidelines E-0 and E-1 as revised by the W Owners' Group letters dated October 16, 1979, October 31, 1979, and November 2, 1979 are acceptable for plants having high head safety injection pumps similar to the 412 standard plant, provided that licensees implement the requirements noted above when they develop their procedures. For the case of 4-loop, 3-loop, and 2-loop plants with nominal 1400 psi range safety injection pumps, the W Owners' Group has committed to submit revisions to the guidelines for these plants which are similar to those provided for the 412 standard plant in the W Owners' Group letters dated October 31, 1979 and November 2, 1979. Based on this commitment, we find the guidelines for these plants acceptable, pending submission of such revisions, subject to the requirements on individual licensees identified above."

Since the W Owners' Group fulfilled its commitment regarding submittal of the revised guidelines for the nominal 1400 psi safety injection plants, we gave the Owners' Group unconditional approval of thee guidelines in our letter of December 6, 1979.

Licensees with W-designed operating plants are now proceeding with development of small break LOCA emergency procedures and training of operators based on the approved guidelines.* As part of our audit program, we will examine the procedures of lead plants in several of the classes of W-designed operating plants to assure that they have been developed according to the approved guidelines. We will also check out some of the procedures at a W PWR is simulator.

2.3 Findings

(1) Before the TMI-2 accident, little attention was paid to operational procedures, especially to emergency procedures and their relationship to the supporting safety analyses. Unambiguous diagnostics and proper precautions and prohibitions were not always considered in the development of procedures.

(2) The NSSS vendor usually does not check a customer's procedures to determine whether the vendor's operational guidelines have been properly incorporated into the plant procedures.

(3) Emergency operating procedures currently in use at operating plants have evolved on an "event-specific" basis. Symptom-based emergency procedures, which are categorized according to general plant symptoms and include the essential features of several separate existing associated procedures, could make use of the fact that the initial operator responses to the associated events are similar.

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*The guidelines were modified by the W Owners' Group in the letter, Cordell Reed to D. F. Ross, Jr., dated December 21, 1979. By letter D. F. Ross, Jr., to Cordell Reed, dated December 27, 1979, we approved these modifications.

The principal advantages of symptom-based emergency procedures over event-specific procedures are (1) the procedures, as a whole, would be simplified significantly and would, therefore, not require the operator to make a detailed diagnosis of the plant conditions prior to consulting an emergency procedure, (2) the total number of emergency procedures with which the operator would have to contend during an emergency would be reduced significantly, and (3) such an exercise would necessitate that licensees look again at their emergency procedures in a more integrated manner.

We believe that the aforementioned advantages would contribute significantly to the operator's ability to maintain the plant in a safe condition in the event of an emergency.

Recommendations

(1) The NRC should become more involved in the review of procedures, including their correlation with the assumptions made in the supporting safety analyses. The procedures should include recognition of the event, precautions, actions, and prohibited actions.

(2) Independent of the NRC review of procedures, the NSS vendor should confirm that the vendor's operational guidelines have been properly incorporated into the customer-licensee's plant operating procedures. Any exceptions which the customer-licensee may have taken to the vendor's guidelines should be documented with appropriate justification. Copies of the correspondence regarding such exceptions should be transmitted to the NRC for information.

(3) Licensees whose emergency procedures have been developed on an event-specific basis should restructure and reformat them on a symptom basis.

3. HUMAN FACTORS

3.1 Finding (From NUREG-0560)

The operator must understand his responsibilities during abnormal and emergency conditions. The design basis for the plant has provided that, in the event of emergencies, suitable actions will be automatically initiated by the safety systems. The operator's initial responsibility is to monitor the parameters of interest and verify that appropriate safety systems have been actuated. If the appropriate actuations have not occurred, the operator must intercede and perform the actions necessary to implement them. The operator is trained to believe his instrumentation. However, he must be trained not to rely on a single instrument, since any single indication may be erroneous or misleading under certain conditions. The reason for this precaution was clearly illustrated at TMI-2, where operator attention was focused on the pressurizer level indication. In virtually all situations, other instrumentation can be used to corroborate or refute the validity of a given instrument.

3.2 Recommendation (From NUREG-0560)

The operator should monitor the control board and evaluate all parameters of concern by appropriate checking of other instrumentation. He must perform this cross-check to verify instrument display. If he has additional manual actions to perform, he may reduce his observations on other system parameters that may lead him to "tunnel vision". This recommendation should be implemented in operator training programs.

APPENDIX X
PLANT-SPECIFIC
AUXILIARY FEEDWATER SYSTEMS

1. INTRODUCTION

In order to assure that the evaluation of the auxiliary feedwater (AFW) system was based on design and operational information applicable to the "as-built" AFW system for each plant, the staff submitted to each licensee the request for information included as Annex 1 to this appendix. Each licensee was requested to bring the requested information to a meeting with the staff. This information was reviewed and discussed with each licensee at separate meetings for each plant.

The information provided by each licensee was evaluated by a review team consisting of a systems engineer and a systems reliability engineer, with emphasis given to improving AFW system performance reliability, including identification of dominant failure modes and consideration of the potential for human operators to affect AFW system performance and reliability.

2. OPERATING PLANTS WITH WESTINGHOUSE-DESIGNED NUCLEAR STEAM SUPPLY SYSTEMS

The plant specific AFW system evaluation were sent by the NRC staff to the licensees of W plants in the period October-November 1979. Enclosoure 1 of these letters which contained the staff evaluations, are provided in this appendix for reference. The plants and the applicable staff letters are provided in subsequent section of the Appendix.

Plant Name	Staff Letter
Beaver Valley 1	Oct. 11, 1979
D. C. Cook 1 & 2	Oct. 30, 1979
Farley Unit 1	Oct. 13, 1979
Ginna	Oct. 22, 1979
Haddam Neck	Oct. 11, 1979
H. B. Robinson 2	Sept. 21, 1979
Indian Point 2	Nov. 7, 1979
Indian Point 3	Nov. 7, 1979
Kewaunee	Sept. 21, 1979
North Anna 1	Sept. 28, 1979
Prairie Island 1 & 2	Oct. 16, 1979
Point Beach 1 & 2	Sept. 21, 1979
Salem 1	Sept. 21, 1979
San Onofre 1	Nov. 15, 1979
Surry 1 & 2	Sept. 25, 1979
Trojan	Oct. 3, 1979
Turkey Point 3 & 4	Oct. 16, 1979
Yankee Rowe	Nov. 9, 1979
Zion 1 & 2	Sept. 18, 1979

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3. BEAVER VALLEY UNIT 1 AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

X (W)

BEAVER VALLEY UNIT 1
AUXILIARY FEEDWATER SYSTEM

X.1.1 System Description

X.1.1.1 Configuration, Overall Design

A simplified flow diagram of the Beaver Valley Plant, Unit No. 1, Auxiliary Feedwater System (AFWS) is shown in Figure 1. The AFWS consists of one turbine driven pump (700 gpm @ 2696 ft head), and two motor driven pumps (350 gpm @ 2696 ft head). The pump discharge headers are connected to permit auxiliary feedwater delivery to any one or all three steam generators by any AFW pump. The licensee states that for normal and transient plant operation, including loss of main feedwater flow, only one pump is required to cool the plant down to the condition where the RHR system can be put into operation to continue safe plant shutdown. However, in the event of an unisolable main steam or main feed line break, either one turbine-driven AFW pump or both motor-driven pumps are required to prevent dryout of the steam generators.

The primary water supply of the AFWS is maintained in a 140,000 gallon seismic Category I, primary plant demineralized water storage tank (DWST). The tank is reserved strictly for the AFWS pump usage. The reserved water inventory is sufficient to maintain the plant at hot standby condition for 8 hours following a reactor trip. Low water level in the DWST will alarm and annunciate in the main control room. The secondary water supply is the seismic Category I river water system with an additional backup source from the fire protection system.

X.1.1.2 Components - Design, Classification

All pumps, valves, piping, instrumentation and controls associated with the auxiliary feedwater system are designed to seismic Category I requirements.

The primary water source (Demineralized Water Storage Tank) and the secondary water source (River Water System) are also designed to seismic Category I requirements. The additional backup water source from the fire protection system is not designed to seismic Category I requirements.

X.1.1.3 Power Sources

The turbine driven pump is supplied with steam from each steam generator outlet header upstream of the main steam isolation valve (MSIV) and exhausts to the atmosphere. The motor driven pumps receive power from the 4160 V AC vital buses. In the event of a loss of offsite power, the pumps are powered by the Division A and B emergency diesel generators, respectively.

X.1.1.4 Instrumentation and Controls

X.1.1.4.1 Controls

The control of auxiliary feedwater flow and steam generator water level is accomplished from the main control room by manually operated control valves. These valves can also be manually operated from the local shutdown control panel if the control room is not accessible. All manually operated valves in the main flow path of the AFWS are either "lock-opened" or "lock-closed" in their normal position. The motor operated valves fail in their "as is" position.

X.1.1.4.2 Information Available to Operator

The important information available to the operation includes AFW discharge header pressure, AFW flow to each steam generator, DWST water level, steam generator water level, steam pressure to turbine driven AFW pump and control valve position indicators. Additional information available is in the following instrument list:

SPECIFIC INSTRUMENTATION AND CONTROL

Flow

Auxiliary feed flow to 1A (B, C) Steam Generator

FI-FW-100A (B, C)

Readout location: Vertical Board - Section C

FI-FW-100A1 (B, C)

Readout location: Emergency Shutdown Panel

AUX STEAM GEN FEED PUMP AUTO START-STOP

PRI PLNT DEMIN WTR STRGE TNK LVL H-L CH 1

PRI PLNT DEMIN WTR STRGE TNK LVL H-L CH 2

1/3 STM GEN 1A HI-HI LEVEL

1/3 STM GEN 1A LOW LOW WATER LEVEL

STM GEN 1A LOW WTR LEVEL CH 1

STM GEN 1A LOW WTR LEVEL CH 2

1/3 STM GEN 1B HI-HI LEVEL

1/3 STEAM GEN 1B LOW LOW WATER LEVEL

STM GEN 1B LOW WTR LEVEL CH 1

STM GEN 1B LOW WTR LEVEL CH 2

1/3 STM GEN 1C HI-HI LEVEL

1/3 STM GEN 1C LOW LOW WTR LEVEL

STM GEN 1C LOW WATER LEVEL CH 1

STM GEN 1C LOW WATER LEVEL CH 2

AUX STM GEN FEED PUMP 3A START-STOP

AUX STM GEN FEED PUMP 3B START-STOP

AUX STM GEN FD PP 3A MTR INBD BRG TEMP

AUX STM GEN FD PP 3B MTR INBD BRG TEMP

AUX STM GEN FD PP 3A MTR OUTBD BRG TEMP

AUX STM GEN FD PP 3B MTR OUTBD BRG TEMP

X.1.1.4.3 Initiating Signals for Automatic Operation

The AFWS is automatically initiated. It can also be started manually from the main control room. In addition, the pumps can be manually started from the local shutdown control panel. The automatic initiating signals are as follows.

1) Turbine Driven Pump
    a) 1/3 Steam Generator Lo-Lo Level (1 out of 3 channel logic)
    b) Under Voltage

2) Motor Driven Pumps
    a) 2/3 Steam Generator Lo-Lo Level
    b) Both Main Feed Pumps Trip
    c) Safety Injection Signal
    d) Turbine Driven AFW pump low discharge pressure consistent with a start signal on turbine driven pump.
    e) Loss of offsite power

X.1.1.5 Testing

The systems are tested periodically in accordance with technical specification requirements. The frequency of periodic testing is 31 days. In addition, the particular system is tested in accordance with the technical specification after performing system maintenance. The systems are tested using the recirculating lines, with various plant parameters noted (suction and discharge pressures, etc). The instrumentation system is checked periodically, in accordance with the technical specifications, on a per shift, monthly or refueling time frame basis.

X.1:1.6 Technical Specifications

A review of the technical specifications indicated that these specifications cover limiting conditions of operation (LCO) and periodic surveillance testing consistent with current standard Technical Specifications.

X.1.2 Reliability Evaluation

X.1.2.1 Dominant Failure Modes

The following failure modes were found to dominate the demand unavailability of the Beaver Valley Unit 1 AFWS.

Loss of Feedwater (LOFW) with Offsite AC Available

The dominant failure mode ($\geq$90%) for this transient event was assessed to be those possible coupled human errors in testing, i.e., leaving two or more of the manual block valves closed in the discharge side of the pumps while performing the type of pump flow testing required by the Technical Specifications.

The licensee has recognized this possible common mode error and is planning to chain lock all manual valves into their correct alignment state. Further, the licensee will, in the future, stagger his pump test program such that no more than one of the three pumps will be tested in any one shift. Considering implementation of these procedures, the overall availability of the Beaver Valley AFWS design should be improved by roughly a factor of three.

LOFW with only Onsite AC Available

Assessment of the AFWS, given this transient event, indicated that there would be no significant change in the predicted unavailability of the Beaver Valley AFWS. Human error concerning mispositioned block valves in the AFWS discharge remained the dominant failure mode.

LOFW with only DC Available

In this transient event, the Beaver Valley Unit 1 AFWS would be expected to automatically actuate and the human could serve as backup to open any of the valving that failed to electrically respond.

The dominant contributors to AFWS unavailability in this event were:

- allowed test and maintenance outage
- hardware faults (principally the failure of steam turbine pump)

X.1.2.2 Principal Dependencies Identified

The principal dependency identified was the human error (common mode) vunerability associated with manual closure of the AFWS discharge block valves and failure to reopen them.

X.1.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS and plant specific) identifier in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL and plant specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981 or as soon thereafter as is practicable.

X.1.3.1 Short Term

1. Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

   The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

- Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

4. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety grade requirements as indicated in Recommendation GL-5.

   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.

- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

5. Recommendation - The normally closed manually operated suction valves from the river water system to the AFWS should be periodically tested and the position verified. The licensee should propose appropriate Technical Specifications to incorporate these provisions.

6. Recommendation - The strengthened administrative procedures described in Section X.1.2.1 above should be implemented; namely, the locking of manual valves in the correct position and staggered testing of the AFW system pumps. The licensee has advised us that it plans to implement such strengthened procedures before Beaver Valley Unit 1 (currently shut down for reasons unrelated to this AFW system review) returns to power.

7. Recommendation - As shown in Figure 1, the locked block valves in each AFW pump discharge line are aligned so that the combined flow from one motor-driven pump plus one turbine-driven pump is supplied to the steam generators via one AFW header while flow from the remaining motor-driven pump is supplied to the steam generators via the redundant AFW header. As indicated in Section 1.1.1, the licensee states that,in the event of an unisolable main steam or main feed line break, the flow from both motor-driven pumps or from the turbine-driven pump is required to prevent dryout of the steam generators. The licensee should review the present alignment of the AFW pump discharge block valves and modify as necessary to provide the AFW required for normal, transient, and accident conditions.

X.1.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and than restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

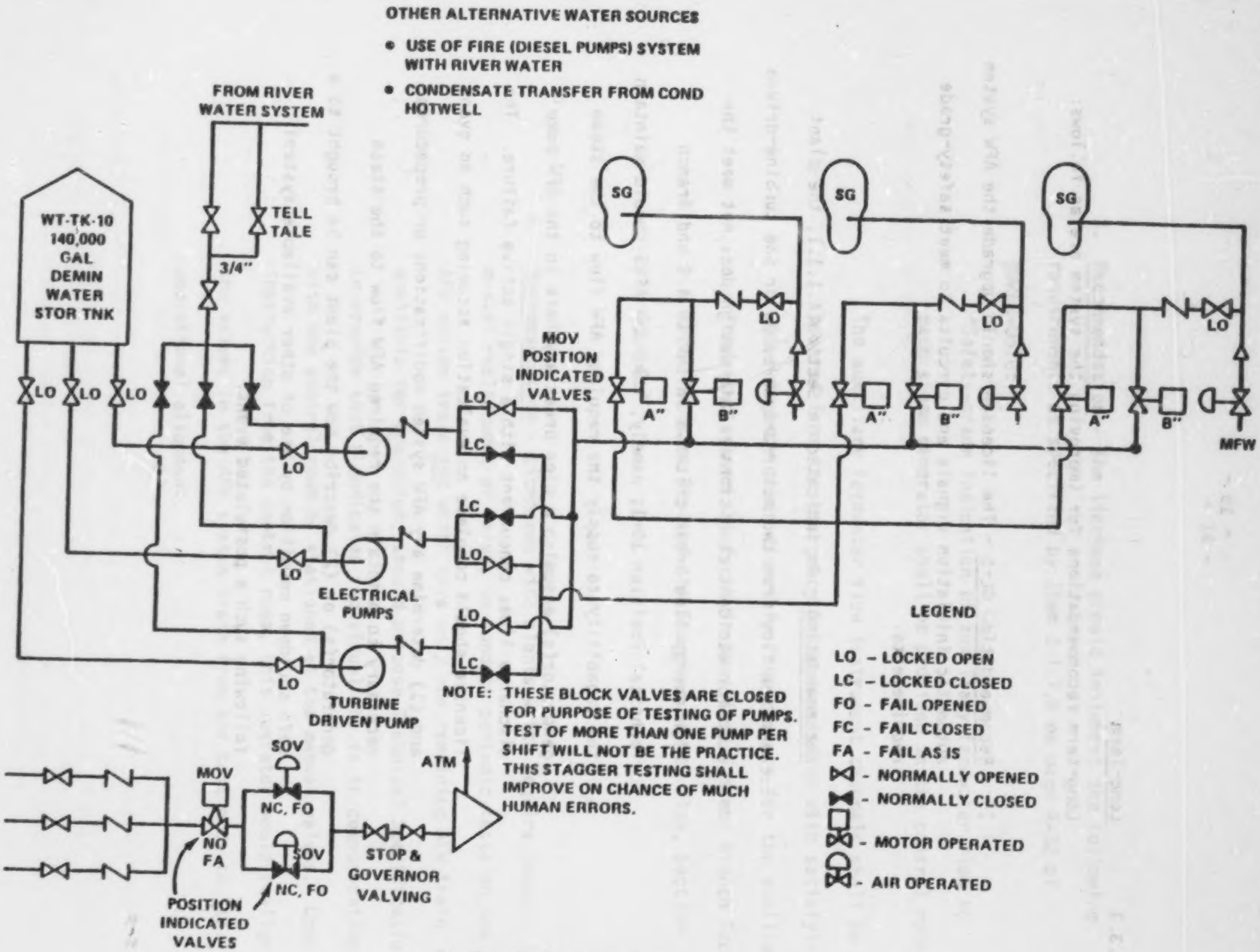
4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.1.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

2. Recommendation - As indicated in Section X.1.1.1, the plant requires flow from two motor-driven pumps or one turbine-driven pump for accident conditions. This design does not meet the high energy line break criteria in SRP 10.4.9 and Branch Technical Position 10-1; namely, that the AFWS should maintain the capability to supply the required AFW flow to the steam generator(s) assuming a pipe break anywhere in the AFW pump discharge lines concurrent with a single active failure. The licensee should complete an evaluation assuming such an event and (1) determine any AFW system modifications or procedures necessary to maintain the required AFW flow to the steam generator(s) or (2) describe how the plant can be brought to a safe shutdown condition by use of other available systems following such a postulated event.


OTHER ALTERNATIVE WATER SOURCES
• USE OF FIRE (DIESEL PUMPS) SYSTEM WITH RIVER WATER
• CONDENSATE TRANSFER FROM COND HOTWELL
FROM RIVER WATER SYSTEM
TELL TALE
3/4"
WT-TK-10 140,000 GAL DEMIN WATER STOR TNK
LO
LO
LO
LO
LO
LO
LO
LC
LC
LO
LO
LC
ELECTRICAL PUMPS
TURBINE DRIVEN PUMP
MOV POSITION INDICATED VALVES
SG
SG
SG
A"
B"
A"
B"
A"
B"
LO
LO
LO
MFW
MOV
SOV
NC, FO
NO FA
SOV
NC, FO
POSITION INDICATED VALVES
STOP & GOVERNOR VALVING
ATM
LEGEND
LO - LOCKED OPEN
LC - LOCKED CLOSED
FO - FAIL OPENED
FC - FAIL CLOSED
FA - FAIL AS IS
NORMALLY OPENED
NORMALLY CLOSED
MOTOR OPERATED
AIR OPERATED

NOTE: THESE BLOCK VALVES ARE CLOSED FOR PURPOSE OF TESTING OF PUMPS. TEST OF MORE THAN ONE PUMP PER SHIFT WILL NOT BE THE PRACTICE. THIS STAGGER TESTING SHALL IMPROVE ON CHANCE OF MUCH HUMAN ERRORS.

Auxiliary Feedwater System
Beaver Valley, Unit-1

Figure 1.

4. DONALD C. COOK UNITS 1 AND 2 AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

X.2 (W)

DONALD C. COOK UNITS 1 and 2
AUXILIARY FEEDWATER SYSTEM

X.2.1 System Description

X.2.1.1 Configuration and Overall Design

The auxiliary feedwater system (AFWS) is designed to supply water to the steam generators for reactor coolant system sensible and decay heat removal when the main feedwater system is not available. The AFWS for Cook 1 & 2 is utilized in the event of either a malfunction such as loss of offsite power, or an accident, and during certain periods of normal startup and shutdown. The AFWS is automatically actuated under certain transient and accident conditions.

The AFWS is shown in simplified form on Figure 1 attached. The AFWS, consists of a steam turbine driven pump for each unit which supplies AFW flow to 4 steam generators of its associated unit and two cross-connected motor driven pumps each of which supplies flow to 4 steam generators, two in each unit.

The motor driven and turbine driven auxiliary feed pumps of each unit normally take suction from the condensate storage tank associated with that each unit. A cross-tie line connects the condensate storage tanks and auxiliary feed pump suctions of the two units. An air-operated valve which can be controlled from the control room is provided in the cross-tie line. This valve is normally closed and will fail in the closed position. This valve can also be manually operated at a local valve station.

Each condensate storage tank has a capacity of 500,000 gallons of which 175,000 gallons are reserved by Technical Specification for AFW system use. The licensee estimates that this reserve capacity is sufficient for approximately 12 hours of operation and is adequate to bring the unit to RHR operation capability. Each condensate storage tank is located outdoors and is non-seismic Category 1 design. However, the tanks have been analyzed to show that they can withstand the operating basis earthquake (OBE). If neither condensate storage tank is available, the AFW pumps can take suction from the Essential Service Water System (ESWS) through a normally closed motor operated valve and a normally closed manual isolation valve at each auxiliary feed pump suction. The ESWS piping and valves are designed in accordance with B31.1. However, the entire ESWS system is analyzed to withstand the safe shutdown earthquake (SSE).

All manual valves located between the condensate storage tank and the AFW pumps suctions are locked in the open position.

A manual duplex strainer is installed in each auxiliary feed pump suction to prevent pump damage from debris and/or scale in the water. Also automatic backwash duplex trainers are installed in the ESWS pump discharge lines.

The motor driven AFW pumps supply two steam generators in each unit (i.e., the Unit 1 motor driven pump supplies steam generators No. 2 and No. 3 of Unit 1 and steam generators No. 1 and No. 4 of Unit 2 and the Unit 2 motor driven pump supplies steam generators No. 2 and No. 3 of Unit 2 and steam generators No. 1 and No. 4 of Unit 1.) Each of the motor driven pump supply lines to the steam generators has a normally closed motor-operated valve for flow control and isolation. On loss of power these valves fail AS-IS.

The motor driven AFW pumps are sized to prevent actuation of the pressurizer safety or relief valves in the event of loss of all main feedwater supply in conjunction with loss of power to the reactor coolant pump buses. Each motor driven AFW pump has a capacity of 450 gpm with a TDH of 2714 feet. These pumps are powered from separate emergency electrical buses. The capacity of the two pumps (900 gpm) is sufficient to maintain the level in 4 steam generators above the lower limit of the wide range level indicator.

The turbine driven AFW pumps meet the same criteria as the motor driven AFW pumps except their capacity is 900 gpm with a TDH of 2714 feet.

Steam to each of the turbine driven AFW pumps is supplied from its associated Units' No. 2 and No. 3 steam generators taken upstream of the main steam isolation valves. Each of the turbine driven AFW supply lines to the steam generators has a normally open motor-operated valve for flow control and isolation.

The discharge pipe header and individual supply lines to each steam generator for the motor driven and turbine driven auxiliary feedwater pumps are designed to seismic Category 1 requirements, AEPSC quality level 4 which is equivalent to ASME Class II.

Each AFW pump is provided with an emergency leakoff line and a test line. The emergency leakoff line ensures that a minimum flow through the pump is maintained to prevent pump overheating and possible damage.

Upon automatic startup of the motor driven auxiliary feed pumps, the steam generator blowdown valves and pump test line close. The motor driven valves in the pump discharge lines to the unaffected unit remain closed while the valves to the affected unit open automatically.

A high flow rate through the motor driven or turbine driven pump causes the associated pump's motor-operated isolation valves to the steam generators to automatically close to an intermediate position. The valves may then be operated as necessary from the control room.

X.2.1.2 Component Design Classification

1. The condensate storage tanks are non-seismic but have been analyzed to withstand the OBE.
2. The suction piping and valves from the condensate storage tanks to the AFW pumps are designed to B31.1 with quality control to

the requirement of B31.7. The suction piping is analyzed to withstand the SSE.

3. The ESWS (alternate AFW supply) piping, valves and components are designed to B31.1 requirements. However, the ESWS is analyzed to withstand the SSE.

4. The turbine driven pumps and motor driven pumps are designed to seismic Category I requirements.

5. The turbine driven pump discharge header and steam generator supply lines and motor driven pump discharge header and supply lines associated with each Unit are designed to seismic Category 1, AEPSC quality level 4 which is equivalent to ASME Class II.

6. Each motor driven and turbine driven AFW pump is located in a separate seismic Category I enclosure and protected from tornado missiles.

7. Motors, cables and other electrical components required for the AFW system operation are Class 1E.

X.2.1.3 Power Sources

Each Unit has two class 1E power system trains A and B. Each power train contains a 250 V DC Station battery, 4 KV diesel generator and power distribution system.

Each unit's turbine driven AFW pump and associated valves are powered from power train B of its own unit.

Each motor driven AFW pump and associated support system is powered from the A power train of its associated unit.

The motor driven AFW pump discharge valves are powered from the A power train of the unit served, i.e., Unit 1 motor driven AFW pump supply valves to Unit 1 steam generators are powered from train A Unit 1; Unit 1 motor driven AFW pump supply valves to Unit 2 steam generators are powered from train A Unit 2; Unit 2 motor driven AFW pump supply valves to Unit 2 steam generators are powered from train A Unit 2; Unit 2 motor driven AFW pump supply valves to Unit 1 steam generators are powered from train A Unit 2.

The AFW system as presently installed meets redundancy requirements, however, it is dependent on both AC and DC power for automatic operation.

Intended modifications (currently in progress) to the turbine driven AFW pumps will remove the turbine driven AFW pumps AC power dependence for automatic system initiation.

X.2.1.4 Instrumentation and Controls

The instrumentation and control power is supplied form the 120 V AC vital bus system. There are four vital buses, each supplied by an inverter receiving power either from the 600 V AC Class 1E auxiliary buses or the 250 V DC power system. The motor driven pump breaker controls are powered from the Class 1E 250 V DC power system. The Class 1E station batteries are maintained at full charge by battery chargers supplied from the Class 1E auxiliary buses.

X.2.1.3.1 Controls

Controls for the AFW pumps and their associated valves are located in the control room of the unit with which the pump is associated, and are duplicated at the hot shutdown panel and other unit control panels.

a) FMO-211, -221, 231, and -241 (Unit 1) or (Unit 2) are the Steam generator supply valves from the turbine driven auxiliary feed pump (TDAFP). These 4-inch motor operated (Globe type) valves are normally open, but each may be closed by the control room operator in the event of a feedwater or steam line break at the steam generator with which it is associated. They also may be throttled to regulate steam generator level. In the event of a steam line break and rapid depressurization of a steam generator, or upon detection of a high flow at the TDAFP, these valves are automatically driven to an intermediate position to prevent pump runout. On loss of power, the above valves fail AS-IS.

b) FMO-212, -222, -232 and -242 (Unit 1) or (Unit 2) are the steam generator supply valves from the motor-driven auxiliary feed pumps (MDAFP). These 4-inch motor operated (Globe type) valves are normally closed and are opened and/or throttled as described in (a) above. These valves open automatically, as a result of any of the signals which require MDAFP start for that unit. The steam generator supply valves in the other unit will get a signal to close. On loss of power, these valves fail AS-IS.

c) WMO-753, and -754 are the essential service water (EWS) supply valves to the turbine driven and motor-driven auxiliary feed pumps. These 4-inch motor operated (Butterfly type) valves are normally closed, and except for testing under closely controlled conditions, are opened by the control room operator only if water is unavailable from the condensate storage tanks. In addition to the above normally closed MOVs, the ESW supply to the AFW pumps contains normally closed, manually operated, butterfly valves ESW-109, -115, -145 and -240. These valves must be locally opened in the event ESW is required. Operation of these valves can be achieved in less than 10 minutes. On loss of power, the motor operated valves fail AS-IS.

d) MCM-221 and -231 are the steam supply isolation valves to auxiliary feed pump turbines. These 4-inch motor operated gate valves are normally open, allowing steam pressure to be available up to the trip and throttle (T&T) valve at each turbine. The motor operated steam isolation valves MCM-221 and -231 can be opened or closed from the control room and on loss of power they fail AS-IS. The T&T valve opens automatically when the turbine driven AFW pump receives a start signal; however, it is AC-powered and fails AS-IS.

e) FRV-257 and -258 are the emergency leakoff valves for motor-driven and turbine driven auxiliary feed pumps. The valves are 1-inch, air operated diaphragm, globe type, normally open and spring actuated to fail open on loss of air pressure. The valves are automatically modulated (open or closed) when feedwater flow rate to the steam generators or through the test line is below or above the required minimum pump leakoff flow rate setpoint.

f) CRV-51 is the condensate storage tank cross-tie valve. This 8-inch, air operated diaphragm, globe type, valve is normally closed and is spring actuated to fail closed on loss of air pressure. The valve connects Unit 1 and Unit 2 Condensate storage tanks. This valve is opened by energizing a solenoid valve in the air supply line. Controls for remote operation of this valve are located in the control room. Power to energize the solenoid valve is supplied from the normal station AC power supply.

Opening this valve and manually realigning others in the AFW systems pump suctions permits the AFW systems of both Units to draw condensate from one tank in the event the other tank is not available.

X.2.1.4.2 Information Available to the Operator

The following is information available to the operator on the Main Control Board or on the Hot Shutdown Panel:

1. Flow, gpm to each steam generator
2. Steam generator levels
3. Breaker position (motor driven pump)
4. Motor current and voltage (motor driven pump)
5. Motor operated valve status lights from limit switches
6. Steam pressure to auxiliary feed turbine (as steam generator pressure)
7. Pump discharge pressure
8. Condensate storage tank level
9. Turbine driven auxiliary feed pump speed control
10. Operational alarms and annunciations shown on the attached Table A and B

X.2.1.4.3 Initiation Signals for Automatic Operation

The turbine driven and motor driven auxiliary feedwater pump start signals are listed below.

A. Turbine Driven

The following signals in one unit will start that unit's turbine driven auxiliary feed pump:

(1) Low-Low level in any 2 of 4 steam generators (possible loss feedwater or steam line break)

(2) Reactor coolant pumps bus undervoltage (anticipation of loss of offsite power)

(3) Manually

B. Motor Driven

The following signals in either unit will start both motor driven auxiliary feedwater pumps:

(1) Low-Low level in any steam generator (possible loss of feedwater or main steam line break)

(2) Trip of main feed pumps in either unit

(3) Any safety injection signal derived from Reactor Protection System and/or containment pressure - High at 1.2 psi.

(4) Loss of offsite power* (Pump is sequenced ON when emergency diesel generator is energizing safeguards bus)

(5) Manually

*Note

There is a delay of <60 seconds in starting the motor driven pump. The reason for this delay is to limit the loads during emergency diesel generator loading.

X.2.1.5 Testing

The AFW system and components are tested in accordance with Technical Specification requirements. The frequency of periodic testing of the pumps is 31 days. In addition, the particular system is tested in accordance with the Technical Specification after performing system maintenance. The systems are also tested using the pump recirculation lines test lines with various plant parameters noted (as called out by ASME Section XI). The instrumentation systems are checked in accordance with the technical specifications, on a per shift, monthly or refueling time frame basis.

FRV-255 and -256 are test valves for the motor and turbine-driven auxiliary feed pumps. These normally closed, 3-inch, air-operated, globe type valves are capable of passing approximately the design flow rates for each pump and are used to performance test the pumps on a periodic basis. The valves are diaphragm, spring close type and on loss of air pressure they fail closed. Operating air to each test valve is controlled by a solenoid valve installed in the air supply line. Should test valves FRV-255 and -256 be left in the open position, an automatic start of the auxiliary feed pumps will automatically close the valves.

X.2.1.6 Technical Specifications

A review of the technical specifications indicated that these specifications cover limitng conditions of operation (LCO) and periodic surveillance testing consistent with current standard technical specifications.

X.2.2 Reliability Evaluation

X.2.2.1 Dominant Failure Modes

The D. C. Cook auxiliary feedwater system was analyzed to determine the dominant failure modes under three transient conditions:

(a) LOFW with offsite power available

(b) LOFW with onsite power available

(c) LOFW with only DC power available

Results of the analysis are summarized below.

LOFW with Offsite Power Available

No significant failure modes were identified in the analysis. No significant single or double failures were noted. The most dominant failure modes appear to be triple failures involving maintenance in one of the pumps trains and independent failures in the other.

LOFW with Onsite Power Available

The system was analyzed to determine if the dominant failure modes would be significantly different given loss of offsite power. As in the previous case, the dominant failure modes would appear to involve three independent failures. The most dominant mode would involve maintenance of one pump train and hardware failures in another, and failure of the diesel powering the third.

LOFW with Only DC Power Available

Assuming loss of all AC power, the turbine-driven train must be manually actuated by locally opening the turbine-driven AFW pump trip and throttle valve. Assuming that this action is performed, the system should operate successfully, unless failures were to occur in this train or maintenance was being performed at the time of loss of power. The dominant failure modes for this case appear to be:

(a) operator fails to manually open the steam admission valve;
(b) turbine train unavailable due to maintenance.

It should be noted that the licensee is installing the capability to open the steam admission valves by use of a DC source.

X.2.2.2 Dependencies

No locational or environmental dependencies were identified which could cause common-mode failures of the system. The four pumps for the station are located in separate rooms equipped with adequate drains and protected against pipe whip and missiles. In addition, no common dependencies on AC or DC power were identified.

X.2.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-speci identified in this section represent actions to improve AFW system realiability that should be implemented by January 1,1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL, and plant-specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.2.3.1 Short Term

1. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

- The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-5 - Modifications currently are being implemented to make the turbine driven trains independent of any alternating current power source. The following recommendation should be met in the interim. The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operate would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations

should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedure should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5. Recommendation GS-7 - The licensee should verify that the automatic start AFW signals and associated circuitry are safety grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.2.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and a low level alarm in the control room for the AFW system primary water supply to allow the operator anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety related equipment in the room.

3. Recommendation - The licensee should implement the following requirements which are identical to Item 2.1.7.b of NUREG-0578:

- Safety-grade indication of AFW flow to each steam generator should be provided in the control room.

The auxiliary feedwater flow instrument channels should be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

4. Recommendation - Licensees with plants require local manual realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.2.3.3 Long Term

Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-3 - The licensee is currently performing modifications to make the turbine driven train independent of A-C power sources. The following recommendation should be met when these modifications are complete. At least one AFW system pump and its associated flow path and essential instrumentation

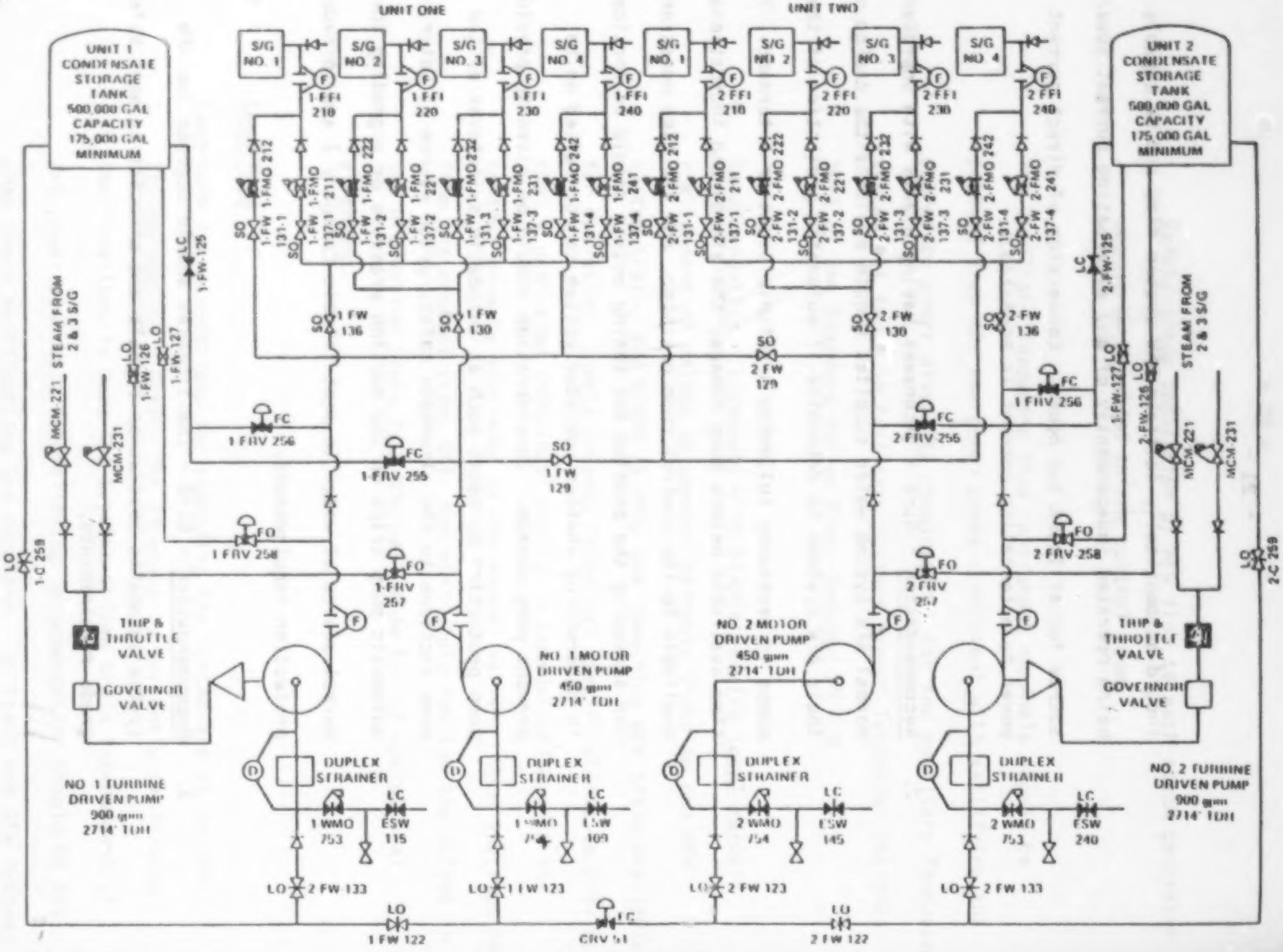
should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

2. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

3. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

NOTE
LEGEND SEE PAGE 2

FIGURE 1
(PAGE 1 OF 2)


UNIT ONE
UNIT TWO
S/G NO. 1
S/G NO. 2
S/G NO. 3
S/G NO. 4
1 FFI 210
1 FFI 220
1 FFI 230
1 FFI 240
2 FFI 210
2 FFI 220
2 FFI 230
2 FFI 240
1-FW 1-FMO 212
1-FW 1-FMO 211
131-1
137-1
1-FW 1-FMO 222
1-FW 1-FMO 221
131-2
137-2
1-FW 1-FMO 232
1-FW 1-FMO 231
131-3
137-3
1-FW 1-FMO 242
1-FW 1-FMO 241
131-4
137-4
2-FW 2-FMO 212
2-FW 2-FMO 211
2-FW 2-FMO 222
2-FW 2-FMO 221
2-FW 2-FMO 232
2-FW 2-FMO 231
2-FW 2-FMO 242
2-FW 2-FMO 241
SO
UNIT 1 CONDENSATE STORAGE TANK 500,000 GAL CAPACITY 175,000 GAL MINIMUM
UNIT 2 CONDENSATE STORAGE TANK 500,000 GAL CAPACITY 175,000 GAL MINIMUM
LC 1-FW-125
LC 2-FW-125
LO 1-FW 126
LO 1-FW-127
LO 2-FW-126
LO 2-FW-127
SO 1 FW 136
SO 1 FW 130
SO 2 FW 130
SO 2 FW 136
SO 2 FW 129
SO 1 FW 129
FC 1 FRV 256
FC 1 FRV 255
FC 2 FRV 256
FC 1 FRV 256
FO 1 FRV 258
FO 1-FRV 257
FO 2 FRV 257
FO 2 FRV 258
STEAM FROM 2 & 3 S/G
MCM-221
MCM-231
MCM 221
MCM 231
LO 1-C 258
LO 2-C 259
TRIP & THROTTLE VALVE
GOVERNOR VALVE
F
NO. 1 MOTOR DRIVEN PUMP 450 gpm 2714' TDH
NO. 2 MOTOR DRIVEN PUMP 450 gpm 2714' TDH
NO. 1 TURBINE DRIVEN PUMP 900 gpm 2714' TDH
NO. 2 TURBINE DRIVEN PUMP 900 gpm 2714' TDH
D
DUPLEX STRAINER
1 WMO 753
LC ESW 115
1 WMO 754
LC ESW 109
2 WMO 754
LC ESW 145
2 WMO 753
LC ESW 240
LO 2 FW 133
LO 1 FW 123
LO 2 FW 123
LO 2 FW 133
LO 1 FW 122
LC CRV 51
LO 2 FW 122

Auxiliary Feedwater System
D.C. Cook
Figure 1 (Sheet 1 of 2)

LEGEND:

⋈ = MOTOR OPERATED GLOBE VALVE – OPEN
⧓ = MOTOR OPERATED GLOBE VALVE – CLOSED
⋈ = MOTOR OPERATED BUTTERFLY VALVE – OPEN
⧓ = MOTOR OPERATED BUTTERFLY VALVE – CLOSED
⋈ = MOTOR OPERATED GATE VALVE – OPEN
⧓ = MOTOR OPERATED GATE VALVE – CLOSED
⋈ = MANUAL GATE VALVE – OPEN
⧓ = MANUAL GATE VALVE – CLOSED
⋈ = MANUAL GLOBE VALVE – OPEN
⧓ = MANUAL GLOBE VALVE – CLOSED
⋈ = MANUAL BUTTERFLY VALVE – OPEN
⧓ = MANUAL BUTTERFLY VALVE – CLOSED
▷| = CHECK VALVE
FO = FAILS OPEN
FC = FAILS CLOSED
LO = LOCKED OPEN (LOCK AND KEY)
LC = LOCKED CLOSED (LOCK AND KEY)
SO = SEALED OPEN (DETENT)

Auxiliary Feedwater System
D.C. Cook
Figure 1 (Sheet 2 of 2)

5. FARLEY 1 AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

X.3 (W) FARLEY 1
AUXILIARY FEEDWATER SYSTEM

X.3.1 System Description

X.3.1.1 Configuration, Overall Design

The auxiliary feedwater (AFW) system as shown in the attached simplified diagram consists of three pumps (2 motor driven, 1 turbine driven) each of which is normally lined up to feed all three steam generators. The motor driven pump discharges are cross connected through manually operated, locked-open valves upstream of the motor operated isolation valves to each steam generator. The turbine driven pump supplies each steam generator downstream of the motor operated steam generator isolation valves and the auxiliary feedwater control valves. Check valves are provided downstream of the feedwater control valves that will prevent reverse flow through the control valves.

The primary water supply source for the AFW system is a 500,000 gallon capacity condensate storage tank (CST). 150,000 gallons are reserved for decay heat removal in the event of an accident. The licensee states that this reserve capacity is sufficient to maintain the plant at hot standby for two hours and cooldown to conditions that the RHR system can be operated. The CST is normally lined up to supply water to the AFW pumps, through redundant lines (one to the motor driven pumps and one to the turbine driven pump) through locked-open, manually operated isolation valves.

A backup long term source of water supply is provided by redundant service water trains. Two normally closed motor operated valves in series isolate the service water trains from the auxiliary feedwater system. These valves can be operated from the control room (key locked) to initiate service water flow to the AFW system. One SWS train is normally lined up to supply a motor driven pump and a turbine driven pump; the redundant SWS train normally supplies the other motor driven pump. With manual valve operation outside the control room, each SWS train can supply all three AFW pump suctions.

X.3.1.2 Components

All components of the auxiliary feedwater system are designed to Quality Group C, seismic Category I requirements including motor, pumps, piping, valves, and valve operators. The auxiliary feedwater control valves, which are the only normally closed valves in the system flow path, are air operated and DC power controlled. The air system is non-safety grade and the control valves will fail open on loss of air or DC control power.

X.3.1.3 Power Sources

The motor driven pumps are powered from independent Class 1E emergency buses supplied by the diesel generators. All valves in the motor driven trains are A-C motor operated or manual valves with the exception of the auxiliary feedwater control valves which are air operated and controlled by DC power. One air compressor can be powered by the diesel-generators to supply air, but it is not safety-grade. The control valves will fail open for maximum AFW flow on loss of air or DC power.

The turbine driven train can be operated and controlled from the control room, independently of AC power. The steam inlet valves are air operated, fail closed, with a one-hour air accumulator available for valve operation upon loss of air. The accumulator will open the valve and keep it open upon loss of instrument air supply. DC control power is used to actuate valve operation.

The backup service water system supply series isolation valves are powered from the same Class 1E bus that powers the service water system train and motor driven AFW pump train. Therefore a single failure of one bus will not disable both backup service water supplies.

There are two motor operated series isolation valves in the flow paths to each steam generator from the motor driven pumps that are powered from separate Class 1E buses such that a single failure of a bus will not prevent isolation of an affected steam generator following a main steam or feedwater line break.

The circuit breakers for the motor driven pumps require DC control power to operate and energize the AFW motors. These breakers can be manually closed locally without DC power.

X.3.1.4 Instrumentation and Controls

X.3.1.4.1 Controls

Steam generator level is controlled manually from the control room. Flow to the steam generators from the motor driven pumps is controlled by

›dulating three flow control valves, one to each steam generator. These valves open for full flow to the steam generators following AFW system initiation. This flow cannot be varied if the system was automatically started by a safety injection signal, until the injection signal has been reset which is 60 seconds after the receipt of the signal.

The flow to each steam generator from the turbine driven pump will normally be controlled from the control room by varying turbine speed. The flow control valves.to each steam generator from the turbine driven pump will normally be kept full open.

Each pump and all motor operated and air operated valves can be operated from the control room and are powered from essential Class 1E buses.

X.3.1.4.2 Information Available to Operator

The control room operator has the following indications and alarms available in the control room.

1. Motor driven Aux Feedwater Pumps
    a. Ammeter
    b. Breaker Status
    c. Monitor Light (Pump Running)
    d. Fault Trip Alarm (overcurrent)
    e. Pump in Local Control Alarm
    f. Breaker Fails to Close Alarm (Loss of Offsite Power & SIAS) indication and alarm

2. Steam Admission Valves Turbine Driven AFW Pump
    a. Valve Status indication
    b. Monitor Light
    c. Valve in Local Control Alarm
    d. Turbine Driven Pump - Fault Alarm - (Overspeed Trip, Steam Valves Closed with Demand Signal)

3. General
    a. Valve Position Indication for all motor and air operated valves
    b. Turbine Speed
    c. Turbine Steam Pressure
    d. Flow to each Steam Generator
    e. Pump Discharge Pressure
    f. Pump Suction Pressure
    g. Condensate Storage Tank Level
    h. Steam Generator Level
    i. Low Suction Pressure Alarm to each pump
    j. Loss of Ventilation Cooler for each motor driven pump room
    l. Hi/Lo Suction flow alarm

X.3.1.4.3 Initiating Signals for Automatic Operation

Motor Driven Pumps

1. Lo-Lo S/G level 2 out of 3 detectors to any one steam generator

2. Both Main Feed Pumps Trip (senses stop valve to turbine driven main feed pumps)

3. Loss of offsite power or two out of three undervoltage condition on respective ESF buses.

4. Safety Injection Signal

Turbine Driven Pump

1. Lo-Lo Steam Generator Level 2 out of 3 detectors to any 2 Steam Generators

2. Undervoltage to any two of three reactor coolant pump buses.

X.3.1.5 Testing

1. The motor driven pumps and the turbine driven pump are tested for operability by recirculation back to the condensate storage tank monthly. Each valve in the auxiliary feedwater system flow path or bypass flow path that is not locked, sealed or otherwise secured in position is verified to be in its correct position at least once per month. Motor operated stop check valves in AFW discharge to each steam generator are verified to be open with the breaker to the valve operators locked open at least once a month.

2. At least once per 18 months during shutdown:
   a. Verify that the motor driven pumps will start upon receipt of the following signals;

(1) Loss of main feedwater pumps

(2) safety injection signal

(3) steam generator water level low-low from one steam generator

(4) loss of offsite power

b. Verify that the steam turbine driven pump starts automatically upon receipt of the following:

(1) .Blackout Signal (undervoltage to RCP buses)

(2) Steam generator low-low water from two steam generators

Valve operability tests are performed quarterly on motor and air operated valves. Stroke tests for these valves are performed quarterly.

X.3.1.6 Technical Specifications

With any one auxiliary feedwater pump inoperable, restore three auxiliary feedwater pumps (2 motor, 1 steam) to operable status within 72 hours or be in Hot Shutdown within next 12 hours. This is in accordance with limiting conditions for operation of Standard Technical Specifications.

X.3.2 Reliability Evaluation

X.3.2.1 Dominant Failure Modes

The dominant failure modes are expressed for three transient situations. Success criterion is the operation of at least one of the three pump trains.

LOFW with Offsite Power Available

The unavailability of the AFWS during this type of transient is dominated by several combinations of three failure elements. These include test and maintenance outages and hardware failures in various combinations and a combination of failures in source lines from the condensate storage tank along with failure of the service water system backup.

Test and maintenance outages of turbine driven pump train and motor driven pump trains are based on monthly pump tests as well as 72 hour allowable maintenance periods for each train. The hardware failures of for the motor driven pump trains include pump failure, in-line valve failures and control signal failure to the pumps.

The hardware failures for the turbine driven pump train include pump failure, in-line valve failures and valve failures in the steam supply lines, including control of steam inlet valves. While the determination of dominant failure contributors is based on systems of this type in general, specific failure data for Farley in its early life shows a series of failure on demand due to trip throttle valve action at the steam inlet to the turbine driven pump. Failures of the supply line valves in the closed position are considered independent human errors.

LOFW with loss of Offsite Power but with Onsite AC Availabl

The conditional unavailability of the AFWS during this type of transient is dominated by the same failure contributors as in the LOFW with Offsite Power Available transient with the addition that failure of one of the two motor driven pump trains can come from potential one train failure of onsite power .

LOFW with loss of all AC, DC Available

Only the steam turbine driven pump train can be operable in this type of situation and failure contributors include test and maintenance and hardware single failure elements. Also included is a failure to manually reset steam inlet valves which are operated by AC derived compressed air.

).3.2.2 Interdependencies

The principal noted dependency is the AC derived compressed air which operates the steam turbine steam inlet valves. Loss of AC and compressed air supply result in eventual bleed-off and fail-closed of the steam inlet valves.

X.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.3.3.1 Short-Term

1. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.
2. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube

oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

3. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:
   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

4. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.

- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

- Testability of the initiation signals and circuits shall be a feature of the design.

- The initiation signals and circuits should be powered from the emergency buses.

- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.3.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level

alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

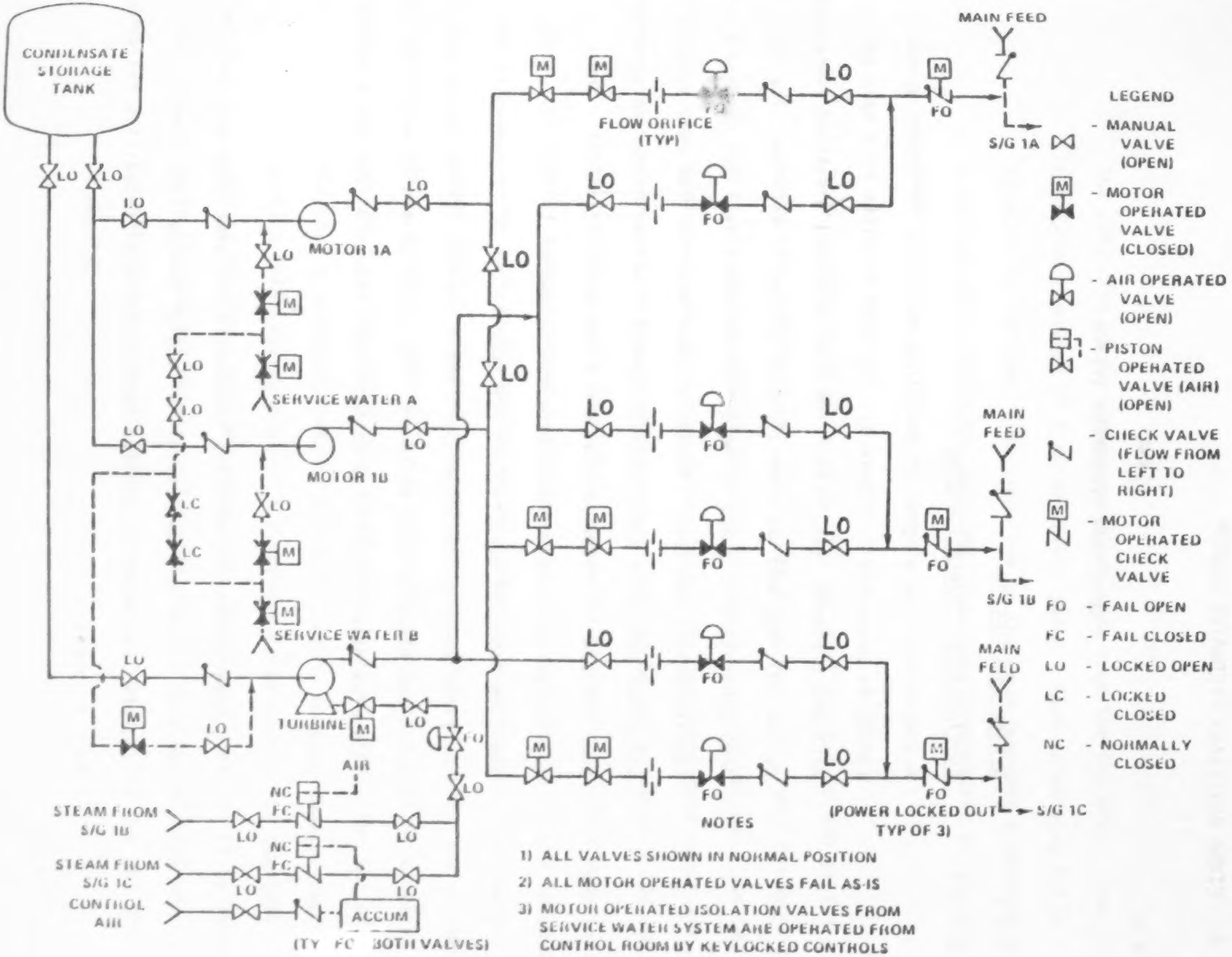
> "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.
>
> The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the contro room, this operator would re-align the valves in the AFW system trai from the test mode to its operational alignment.

X.3.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating curren is acceptable.
2. Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.


CONDENSATE STORAGE TANK
LO
MOTOR 1A
MOTOR 1B
SERVICE WATER A
SERVICE WATER B
LC
TURBINE
AIR
NC
FC
STEAM FROM S/G 1B
STEAM FROM S/G 1C
CONTROL AIR
ACCUM
(TY FC BOTH VALVES)
FLOW ORIFICE (TYP)
FO
MAIN FEED
S/G 1A
S/G 1B
S/G 1C
(POWER LOCKED OUT TYP OF 3)
LEGEND
- MANUAL VALVE (OPEN)
- MOTOR OPERATED VALVE (CLOSED)
- AIR OPERATED VALVE (OPEN)
- PISTON OPERATED VALVE (AIR) (OPEN)
- CHECK VALVE (FLOW FROM LEFT TO RIGHT)
- MOTOR OPERATED CHECK VALVE
FO - FAIL OPEN
FC - FAIL CLOSED
LO - LOCKED OPEN
LC - LOCKED CLOSED
NC - NORMALLY CLOSED

NOTES

1) ALL VALVES SHOWN IN NORMAL POSITION
2) ALL MOTOR OPERATED VALVES FAIL AS-IS
3) MOTOR OPERATED ISOLATION VALVES FROM SERVICE WATER SYSTEM ARE OPERATED FROM CONTROL ROOM BY KEYLOCKED CONTROLS

Auxiliary Feedwater System
Farley 1
Figure 1

6. GINNA AUXILIARY FEEDWATER SYSTEM

X.4 GINNA

AUXILIARY FEEDWATER SYSTEM

X.4.1 System Description

X.4.1.1 Configuration -Overall Design

A simplified flow diagram of the Ginna auxiliary feedwater system (AFWS) is presented in figure 1. The AFWS consists of a main (M) AFWS and a standby (SB) AFWS. The (SB) AFWS was installed subsequent to the (M) AFWS and has recently been placed in service. The (M) AFWS consists of 3 pumps (2 motor-driven pumps, each 200 gpm, and 1 turbine-driven 400 gpm). Normally, each motor-driven pump supplies one steam generator (SG) but, with operator action either motor-driven pump can provide feedwater to both steam generators (SG). The turbine-driven pump normally provides feedwater to both SGs. Only the flow from one motor-driven AFW pump to one SG is needed to cool the plant down to the temperature where the RHR system can be used to bring the plant to safe shutdown. The steam generator would boil dry in approximately 30 minutes without any feedwater flow and a reactor trip.

All three of the (M) AFWS are located in the same room and could be rendered inoperable as a result of a high energy line break. The (SB) AFWS was added to provide independent AFWS capability following such an event.

best copy available

The (SB) AFWS is in a separate plant area from the (M) AFWS. The (SB) AFWS consists of 2 motor-driven pumps. Each motor pump has a capacity of 200 gpm. The pumps are in the same room, but separated by a partial wall. Thus the (SB) AFWS functions independent of the (M) AFWS.

The primary sources of water for the (M) AFWS are two 30,000 gallon condensate storage tanks (CST). The tanks are non-seismic Category I and are cross-connected through locked-open manual operated valves. The (M) AFWS pumps can draw from either tank. The two condensate tanks are connected to the condenser hotwell and can be connected to a 100,000 gallon non-seismic Category I condensate storage tank. The pump that would transfer water from either the condenser hotwell or the 100,000 gallon tank to the 30,000 gallon tanks is powered fron a non-safety grade supply. There is an emergency procedure for connecting to these water sources. Connection to either of these water sources requires operator action, which takes approximately 15 minutes. The (M) AFWS also has a secondary seismic Category I water source; namely, the service water system (SWS). The primary water source for the (SB) AFWS is the SWS. The SWS draws water from Lake Ontario. It is estimated to take approximately 5 minutes to connect the (M) AFWS to the SWS. There is an emergency procedure for connecting the (M) AFWS to the SWS.

X.4.1.2 Components - Design Classification

The (M) AFWS, the (SB) AFWS, and the SWS have a Class I seismic qualification. The primary source (two 30,000 gallon condensate storage tanks) and associated supply lines to the (M) AFWS pumps suction are non-Class I seismic.

X.4.1.3 Power Sources

The main and standby auxiliary feedwater systems are powered from the emergency buses. The two motor-driven pumps, associated valves and lube oil cooling system for the turbine driven pump in the main auxiliary feedwater system receive motive power from two redundant and independent AC emergency buses. The steam admission and water discharge valves and lube oil cooling systems associated with the steam turbine-driven pump in the main auxiliary feedwater system receive power from the electrical divisions indicated in Figure 1. The two motor-driven pumps and valves in the standby auxiliary feedwater system are supplied from redundant and independent AC emergency buses. The (SB) AFWS is interlocked with the (M) AFWS so that both are not simultaneously loaded onto the vital AC buses to prevent overloading the vital buses on loss of offsite power.

X.4.1.4 Instrumentation and Controls

X.4.1.4.1 Controls

Upon loss of the main feedwater system, the (M) AFWS is automatically initiated to supply water to the steam generators. Thereafter, the level in the steam generator is manually controlled from the control room by adjusting valve positions. The (SB) AFWS is manually initiated and manually controlled from the control room by adjusting valve positions.

X.4.1.4.2 Information Available to Operator

System information available to the operator in the control room to assess the performance of the auxiliary feedwater system is as follows:

- Indicating red (open) and green (close) lights associated with each electrical and pneumatic operated valve.
- Steam generator level
- Steam generator pressure
- Auxiliary feedwater flow indication in each of the two water paths to the steam generators as related to the (M) AFWS.
- Auxiliary feedwater flow indication in each of the two water paths to the steam generator as related to the (SB) AFWS.

The (M) AFW pumps are not automatically tripped as a result of low pump suction pressure conditions. This was a potential concern because the non-seismic condensate storage tank supply lines could be severed by a seismic event causing the loss of suction to the (M) AFWS pumps. There is also no alarm or indication in the control room to alert the operator of low suction pressure conditions at AFWS pumps. However the operator does have CST level and pump discharge pressure and flow indication. Further, however, in the event of seismic damage to the (M) AFWS primary water source, the (SB) AFWS would be available since its water souce (SWS) is seismic Category I.

X.4.1.4.3 Initiating Signals for Automatic operation

The steam turbine-driven and motor-driven pumps and corresponding valves in the (M) AFWS are automatically initiated by the following signals:

·Motor-Driven Auxiliary Feedwater Pump A

  ··2/3 lo-lo level in either SG

  ··Both main feedwater pumps trip

  ··Safety injection initiation

Motor-Driven Auxiliary Feedwater Pump B

  ··2/3 lo-lo level in either SG

  ··Both main feedwater pumps trip

  ..Safety injection initiation

·Steam Admission Valve to the Turbine-Driven Auxiliary Feedwater Pump

  ··2/3 lo-lo level in both steam generators

  ··Loss of voltage on both 4 KV buses (non-safety buses)

.Motor and Turbine Driven Pumps Discharge Valves

  ···pump start

The (SB) AFWS is manually initiated.

Both the main and standby auxiliary feedwater systems flow paths to the steam generators are not isolated automatically as a result of a steam or feedwater (main or auxiliary) line break. The isolation is accomplished manually.

.4.1.5 Testing and Technical Specifications

Subsequent to this review, the licensee proposed a Technical Specification revision which provides limiting conditions of operation and periodic testing for both the (M) and (SB) AFWS. These proposed revisions have been revewed by the staff (Systematic Evaluation Program) and found acceptable. The Technical Specification revisions were approved in Amendment 29 to the Ginna operating license (DPR-18) dated August 24, 1979.

.4.2 Reliability Evaluation

.4.2.1 Dominant Failure Modes

LOFW with offsite power available

Failure of operator to throttle pumps and failure of operator to switch to service water supply and failure of operator to actuate the (SB) AFWS.

The condensate storage tanks have 15,000 gallons dedicated to the (M) AFWS. When the system starts, all 3 pumps have the possibility of starting. Their total capacity is 800 gpm. However, only 200 gpm flow to one SG is necessary. To achieve the 200 gpm flow rate, the operator must either throttle or shut off some pumps. If this action is not taken, the CSTs could empty in 20 minutes. The short time interval may not allow the operator time enough to valve in the backup water source from the hotwells and 100,000 gallon tank. A procedure is available; however, it requires operator action outside the control room. The next alternative is to open a service water system valve which is outside the control room. A procedure exists and the licensee estimates 5 minutes to take this action. The final alternative is to valve in from behind the control panel the (SB) AFWS for which procedures exist. If the operator throttles the pumps correctly initially, there should be adequate time and supply to prevent a problem. The licensee estimates the steam generator boil dry time to be approximately 30 minutes which should allow sufficient time to valve in the service water.

LOFW with onsite power available

Same as for LOFW with offsite power available. For this event the condenser hotwell and 100,000 gallon backup condensate storage tank are not available since the transfer pumps are powered from non-vital AC. bus.

LOFW with only DC available

Failure of the turbine pump train.

This is the short term failure. For this condition, the turbine could eventually fail since the AC powered service water pumps are not operating. Thus, there is no water flow to cool the turbine pump lube oil. The CST (assuming 15,000 gal level) could go dry in 40 minutes and also cause failure. The backup sources from the service water and (SB) AFWS are all AC dependent and would not be available. See Recommendations.

X.4.2.2 Principal Dependencies

1. All (M) AFWS pumps are in the same room with high energy piping over-head. However, a postulated high energy line break in this room is mitigated by the installation of the (SB) AFWS in a separate plant area.

2. The DC controlled turbine lube oil pump forces oil through a heat exchanger which depends on the AC powered service water system to cool the oil. In a total loss of AC, the turbine could fail. See Recommendations.

.4.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this se tion involve system design evaluations and/or modifications to improve system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.4.3.1 Short-Term

1. Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

   The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the reaquired initial AFW system flow will not result in plant damage due to water hammer.

2. Recommendation - The plant has AC dependent service water cooling of the lube oil for the turbine driven pump. The

turbine driven feedwater pump has an AC lube oil pump and a DC lube oil pump. These pumps direct the oil through a heat exchanger which depends on the AC powered service water system pumps to cool the oil. In the event of a total loss of AC power, lube oil cooling capability for the turbine-driven pump will be lost due to the loss of AC power to the service water pumps. The turbine-driven pump could cease to function due to the loss of lube oil cooling. The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. Subsequent to this review, the licensee conducted a test to demonstrate that the turbine-driven pump could operate for two hours without lube oil cooling water flow. The test was run for one hour and 45 minutes with the final one hour and 15 minutes of the test with the pump at rated speed,but at 50% of required plant flow. Preliminary test results indicate the pump and turbine bearing temperatures remained within allowable limits. The staff is evaluating these test results to determine if the test data will support a conclusion that the required AFW flow can be provided independent of any AC power source. Until this evaluation is complete, interim emergency procedures should be established which provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump/turbine bearng and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control fo the AFW system is needed. (See Recommendation GL-3 for the longer term resolution of this concern).

3. Recommendation GS-6 The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independetly verify that the valves are properly aligned.

   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

4. Recommendation GS-7 - The licensee should verify that the automatic start (M) AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the (M) AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.

- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

- Testability of the initiation signals and circuits shall be a feature of the design.

- The initiation signals and circuits should be powered from the emergency buses.

- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.4.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW systems designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits wth respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

> "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.
>
> The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for

operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.4.3.3 Long-Term

Long Term recommendations for improving the system are as follows:

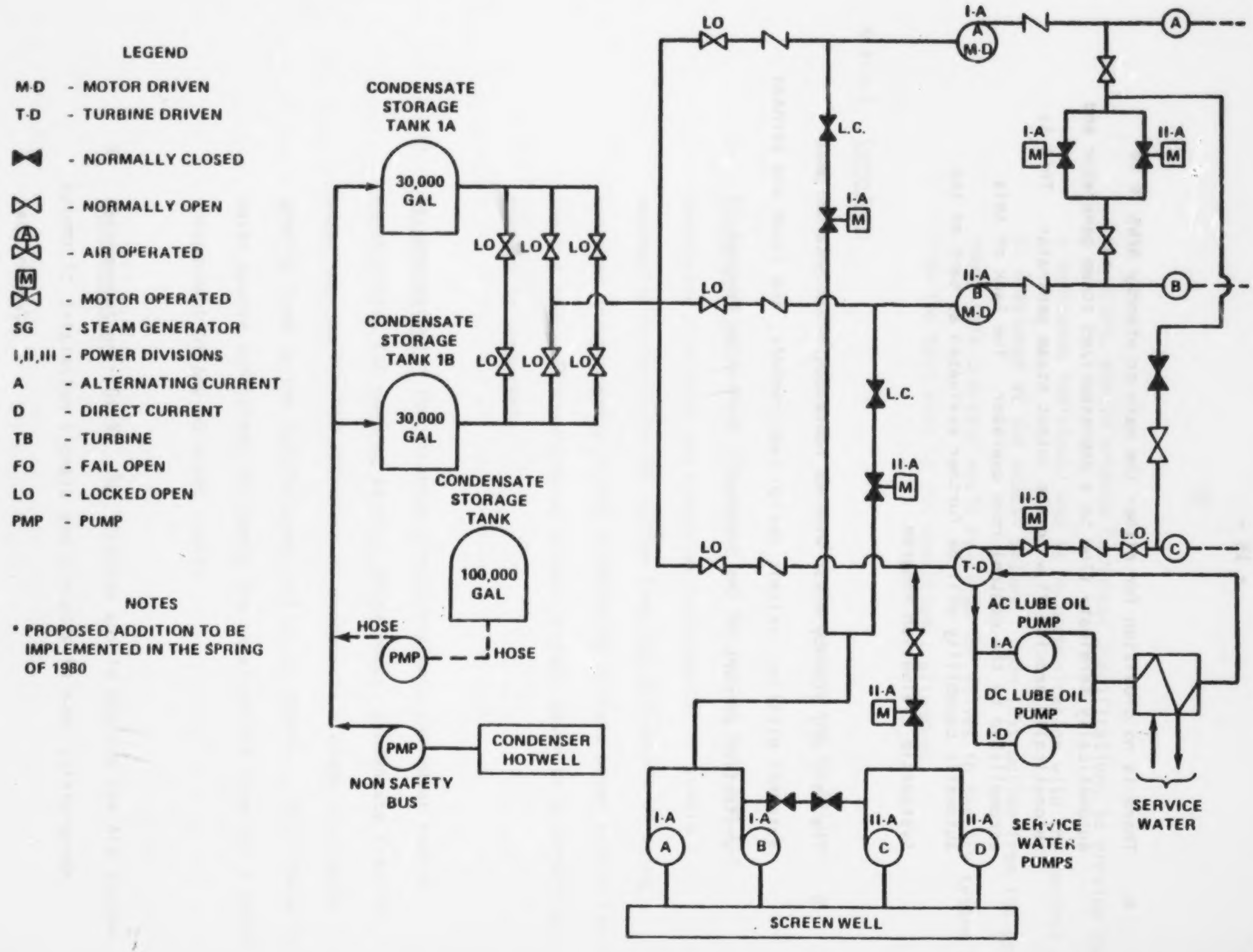
1. Recommendation - GL-3. At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least 2 hours. Conversion of direct current power to alternating current is acceptable.

2. Recommendation - The licensee should evaluate the water source capabilities (AC powered service water pumps, condensate transfer pumps and the limited inventory of condensate storage tank water gravity feed to the turbine pump suction to assure that there is a water source sufficient to supply the required AFW flow for 2 hours independent of any AC power source.

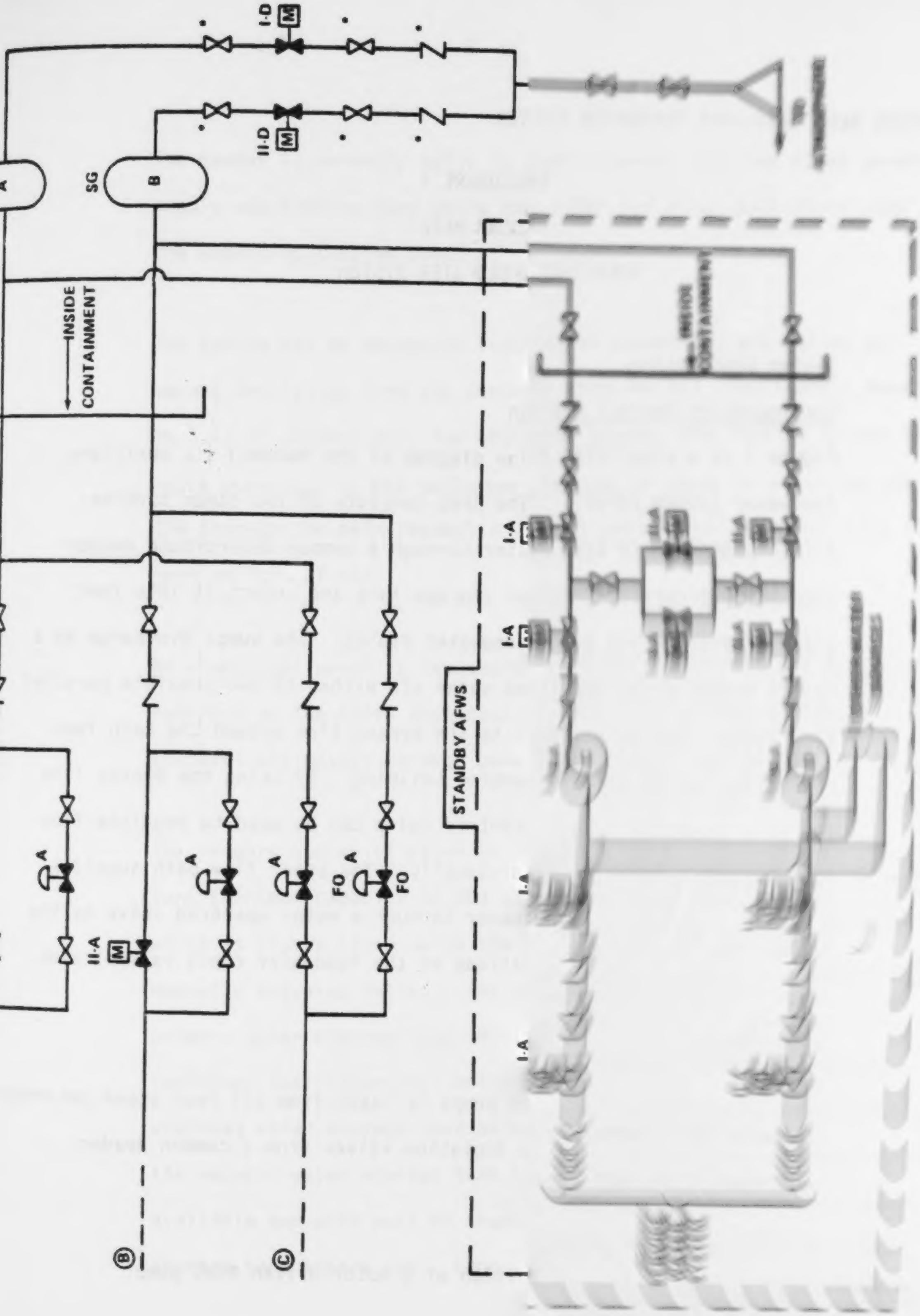
3. Recommendation - GL-5. The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

4. There is no provision for either the main or standby AFWS 's to automatically terminate flow to a depressurized steam generator and automatically provide flow to the intact steam generator. This is accomplished by the control room operator. The lack of this automatic capability will be further evaluated as part of the Systematic Evaluation Program.

5. The main and standby AFWSs will be reevaluated for internal and external missiles, seismic design requirements, and flood and tornado protection as part of the Systematic Evaluation Program.


LEGEND
M-D - MOTOR DRIVEN
T-D - TURBINE DRIVEN
- NORMALLY CLOSED
- NORMALLY OPEN
- AIR OPERATED
- MOTOR OPERATED
SG - STEAM GENERATOR
I,II,III - POWER DIVISIONS
A - ALTERNATING CURRENT
D - DIRECT CURRENT
TB - TURBINE
FO - FAIL OPEN
LO - LOCKED OPEN
PMP - PUMP
NOTES
* PROPOSED ADDITION TO BE IMPLEMENTED IN THE SPRING OF 1980
CONDENSATE STORAGE TANK 1A
30,000 GAL
CONDENSATE STORAGE TANK 1B
30,000 GAL
CONDENSATE STORAGE TANK
100,000 GAL
HOSE
PMP
HOSE
PMP
NON SAFETY BUS
CONDENSER HOTWELL
LO
L.C.
I-A
M
II-A
I-A
A
M-D
II-A
B
M-D
T-D
II-D
L.O.
A
B
C
AC LUBE OIL PUMP
DC LUBE OIL PUMP
I-D
SERVICE WATER
SERVICE WATER PUMPS
C
D
SCREEN WELL

Auxiliary Feedwater System
Ginna
Figure 1 (Sheet 1)


SG
A
B
I-D
II-D
M
INSIDE
CONTAINMENT
STANDBY AFWS
I-A
II-A
A
FO
B
C

7. HADDAM NECK AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

X.5 (W) HADDAM NECK

AUXILIARY FEEDWATER SYSTEM

X.5.1 System Description

X.5.1.1 Configuration Overall Design

Figure 1 is a simplified flow diagram of the Haddam Neck auxiliary feedwater system (AFWS). The AFWS consists of two steam turbine driven pumps* which take water through a common underground header from the demineralized water storage tank and inject it into four steam generators via main feedwater piping. The pumps discharge to a common header which supplies water via either of two possible parallel flow paths. One path feeds to the bypass line around the main feed regulating valves in the turbine building. By using the bypass line, the main feedwater bypass control valve can be used to regulate flow to the steam generators individually. The other flow path supplies water from the discharge header through a motor operated valve to the main feedwater piping downstream of the feedwater check valve inside containment.

Steam to the turbine driven pumps is taken from all four steam generators upstream of the main steam isolation valves from a common header.

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*[illegible]

The header is normally split in such a manner that two steam generators supply one turbine pump while the other two steam generators supply the remaining turbine pump.

The system has no automatic initiation capability and relies on manual initiation from the control room for all conditions. However, on loss of control air, for whatever reason, the turbine driven pumps would start due to the fail-open feature of steam inlet valves and deliver AFW through the main feedwater bypass control valves which also fail open on loss of air.

No electrical power is necessary to operate these valves because the controls at the panel mechanically initiate or remove control air. (Control air passes through panel via copper tubing.)

The primary source of water is from the demineralized water storage tank (Minimum capacity [illegible] gallons by Technical Specifications) which is always lined up to the pump suction header via locked open manually operated valve. The secondary source of water is the primary water storage tank (Minimum volume of [illegible] gallons by Technical Specifications) which must be transferred to the demin [illegible] water storage tank before use. [illegible] the water treatment system using a well pump, the well pump without use of the water treatment system and a diesel driven fire protection system pump. All water sources must eventually come via the demineralized water storage tank.

X.5.1.2 Components Design Classification

The seismic design and safety classification of components for the Haddam Neck plant are being reviewed as part of the Systematic Evaluation Program. The safety classification and seismic design requirements for the plant as compared to today's requirements are too detailed and complex to provide a meaningful explanation in this report. Refer to the details available as part of SEP for this information. The overall design of the auxiliary feedwater system, including the demineralized water storage tank and primary water storage tank, are considered to be seismic Category I based on the licensee's standards. The adequacy of these seismic criteria are also being evaluated as part of SEP.

[illegible]

[illegible] or [illegible] lve [illegible] ater [illegible] lve, [illegible]

Compressed air is used to operate the steam inlet valves and the main feedwater bypass line control valves. These valves are opened or closed at the control panel by controls that are essentially control valves that control the air pressure from the compressed air header to the valve operators. All valves fail in the open position upon loss of air pressure. The compressed air system includes three air compressors and three air receivers for control air. All of the compressors can be powered by the diesel generators.

The AFW pumps have a self-contained lube oil pumping system (shaft driven) but require service water to cool the lube oil. The service water is supplied on a continuous basis to the lube oil coolers (one service water train to each pump). However, the pumps will start and operate for an unspecified time without cooling water. Subsequent to this review, the licensee indicated it is presently in the process of modifying the AFW system to eliminate the need for service water for the AFW turbine driven pump lube oil coolers. The modification will provide a self-contained bearing oil cooling system for each AFW pump. Water will be drawn from the pump first stage discharge and will circulate through all necessary pump and turbine pump and turbine bearings and will return to the AFW pump suction.

X.5.1.4 Instrumentations and Controls (In Control Room)

X.5.1.4.1 Controls

Steam generator level is controlled manually from the control room by varying turbine speed or throttling the feedwater bypass control valve or a combination of both. When the alternate path to the steam generators is used through the motor operated valve directly to the feedwater piping inside containment, level is controlled by turbine speed control.

Controls for the valves to initiate the auxiliary feedwater system through either of the two flow paths are located in the control room. The controls for the normal flow path through the feedwater bypass line are independent of electrical power.

X.5.1.4.2 Information Available to the Operator

I. Alarms

1. Demineralized water storage tank low level
2. Control air system low pressure alarm
3. Discharge header high temperature alarm (indicates backflow from main feedwater system to discharge header via leaky check valve)
4. Hi/Lo steam generator level alarms

II. Indication

1. Electrical position indication for motor operated isolation valve in an alternate flow path
2. Output pressure of controllers to bypass flow control valves and turbine inlet valves (indirect indication of valve position and turbine speed)
3. Steam pressure at inlet to turbines
4. Discharge pressure from pumps
5. Steam generator level
6. Demineralized water storage tank level and temperature

X.5.1.4.3 Initiating Signals for Automatic Control

Not applicable - manual AFWS initiation

X.5.1.5 Testing

The auxiliary feedwater pumps, steam inlet valves, and controls are tested monthly by isolating pump discharge and starting pump from the control room and checking discharge pressure. This same test is performed following return of system to operation after maintenance. A flow test of the auxiliary feedwater pumps is performed annually.

Valve position is verified monthly and the active valves are cycled quarterly. All valves,active and manual,are cycled annually and the stroke times of the active valves verified.

The controls for all valves are used during valve testing for control operability check.

Technical Specifications

The reactor shall not be critical (except for determination of "just critical" rod position and low power tests at or below 10 percent of full power) unless the following conditions are met:

1. One steam driven auxiliary feedwater pump available
2. A minimum of 50,000 gallons in the demineralizer water storage tank and an additional 80,000 gallons in the primary water storage tank.
3. System piping and valves directly associated with the above components operable.

Licensee is planning to convert to standard Technical Specifications and communications with NRC have been started in this regard. In a letter dated June 1, 1979 in response to Bulletin 79-06A, the licensee submitted a license amendment request proposing more comprehensive technical specifications to further assure the availability of the AFW system. The proposed changes include a requirement that both AFW pumps be operable when the reactor is critical and a provision that limits the time that one AFW pump train can be inoperable. The proposed change is currently under staff review.

X.5.2 Reliability Analysis

X.5.2.1 Dominant Failure Modes

LOFW with Offsite Power Available

The principal dominant failure modes include two single failures associated with human failure. One is a human failure to restore to open, following a maintenance action, the suction line valve from the demineralized storage water tank. The second is the human failure to initiate the AFWS upon evidence of need. The latter contributor is reduced to some extent due to recent NRC Bulletin 79-06A for operator personnel specifically dedicated for AFWS initiation.

Other dominant failure modes include failure to reopen valves in both of two systems, and long term allowable maintenance in one pump system combined with hardware or human failure associated with the other pump system.

LOFW with Loss of Offsite Power

Same as above.

LOFW with Loss of Offsite and Onsite AC

The dominant failure is loss of both pumps due to lack of lube oil cooling from loss of all AC.

X.5.2.2 Interdependencies

The principal interdependencies noted are the common valve in the storage tank line and the AC dependence for cooling of the steam driven pumps.

X.5.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.5.3.1 Short Term

1. Recommendation GS-1 - The licensee should propose* modifications to the Technical Specifications to limit the time period that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

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*As discussed in Section 5.15 the licensee has proposed Technical Specification modifications for AFW system which are currently under review by the staff.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be incorporated into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

3. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

4. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable.* Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed.

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* As noted in Section 5.1.3, the licensee is proceeding with AFW system modifications to provide cooling of the turbine driven AFW pump lube oil which is independent of alternating current power.

5. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedure should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendation GS-8 - The licensee should install a system to automatically initiate AFW system flow. This system need not be safety-grade; however, in the short-term, it should meet the criteria listed below, which are similar to Item 2.1.7a of NUREG-0578. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.
   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

- Testability of the initiation signals and circuits shall be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- Any alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

7. Recommendation -

a. According to Haddam Neck surveillance procedure No. 5.1-13, the monthly operational check of the auxiliary feedwater pumps is currently performed by closing a manual valve in the common discharge header of both pumps, isolating the normal flow path of the auxiliary feedwater system. A parallel flow path is available by manual operation from the control room through motor operated valve MOV-35. The monthly pump test should be performed by isolating the pumps individually such that one pump is always available for normal AFW system operation. When the system is converted to automatic operation, then the existing procedure will have to be changed to individual pump isolation tests to allow automatic initiation.

b. According to Haddam Neck surveillance procedure No. 3.1.14, the annual flow capacity test of the AFW pumps is currently performed either at power or in hot standby. During the test temporary piping is connected to a valved flange in the common discharge header to divert flow away from the normal flow paths and direct it to the yard sewers via the temporary piping. This diverts flow from both AFW pumps while the isolation valve in the flange connection is open. This test should not be conducted when the plant is at power since both AFW pumps' availability is affected.

X.5.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety related equipment in the room.

3. Recommendation - The licensee should implement the following requirements which are specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room."

The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require a local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

3.3 Long Term

Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

Licensees with plants in which the primary AFW system water supply passes through valves in a single flowpath, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

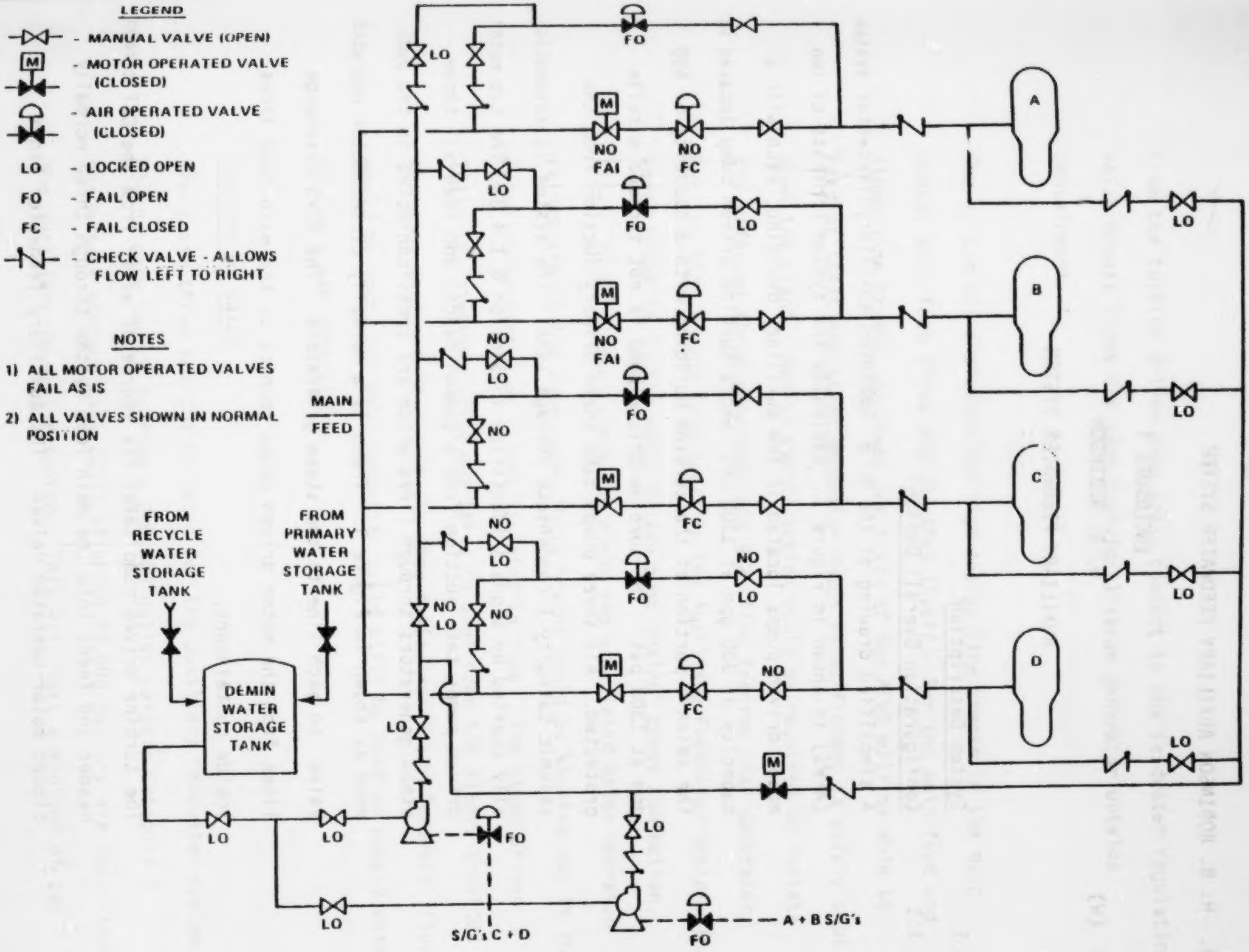
3. Recommendation - There is a common crossconnect line with no isolation valves between the two parallel flow paths to the S/G's. A break in this section cannot be isolated in the present design and the total system would be unavailable. It is recommended that some modifications be made (such as isolation valves) to provide isolation when necessary and assure a means of supplying AFW flow following isolation of such a break. The licensee has begun design plans to add a motor driven pump to the system. The licensee should introduce the flow from this third pump in such a manner that a break in this crossconnect

line will not result in the loss of all pumps. Also the licensee should 1) install the third pump with appropriate valves in the pump discharge line connections to meet the high energy line break criteria in SRP 10.4.9 and Branch Technical Position ASB 10-1; namely, to maintain the capability to supply the required AFW flow to the steam generators with a postulated pipe break anywhere in the AFW pump discharge lines plus a single active failure, or 2) describe how the plant can be brought to a safe shutdown condition by use of other available systems following such a postulated event.

4. The AFW system itself is not designed to withstand a passive failure at all points within the system. A pipe break in a normally pressurized portion of the AFW system can be isolated by operation of manual valves outside the control room. An alternate flow path to all four S/G's would be available following such isolation. The motor driven main feedwater pumps may also be available in this event since no transient should result to cause a loss of non-vital power. For the same reasons, the main feed pumps may also be available following a break in any portion of the AFW system that is not normally pressurized even though the AFW system could be disabled. Further review, including the main feedwater system and time available for operator action, should be conducted to determine if this design has protection equivalent to today's requirements (pipe break

and single active failure). This review is being conducted as a part of Systematic Evaluation Program (SEP).

5. The Systematic Evaluation Program (SEP) will re-evaluate the plant with regard to
   a. internally and externally generated missiles, pipe whip and jet impingement quality and seismic design requirements earthquakes, tornadoes, floods and failure of nonessential systems
   b. the possible need for automatic termination of feedwater flow to a depressurized steam generator and providing flow to the intact steam generator(s). This is accomplished by the control room operator.



LEGEND
- MANUAL VALVE (OPEN)
M - MOTOR OPERATED VALVE (CLOSED)
- AIR OPERATED VALVE (CLOSED)
LO - LOCKED OPEN
FO - FAIL OPEN
FC - FAIL CLOSED
- CHECK VALVE - ALLOWS FLOW LEFT TO RIGHT
NOTES
1) ALL MOTOR OPERATED VALVES FAIL AS IS
2) ALL VALVES SHOWN IN NORMAL POSITION
MAIN FEED
FROM RECYCLE WATER STORAGE TANK
FROM PRIMARY WATER STORAGE TANK
DEMIN WATER STORAGE TANK
A
B
C
D
M
NO
FAI
FC
FO
LO
S/G's C + D
A + B S/G's

Auxiliary Feedwater System
Haddam Neck
Figure 1

8. H. B. ROBINSON AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

X.6 (W)

H. B. ROBINSON
AUXILIARY FEEDWATER SYSTEM

X 6.1 System Description

X 6.1.1 Configuration Overall Design

A simplified drawing of the H. B. Robinson auxiliary feedwater system (AFWS) is shown in Figure 1. Basically the system consists of two motor driven pumps located in the auxiliary building, each with a capacity of 300 gpm at 1300 psi. and a turbine driven pump located in the seismic portion of the turbine building with a capacity of 600 gpm at 1300 psi. The turbine-driven pump is not tornado missile protected. All three pumps take their primary suction from the seismic Category I condensate storage tank. The system is automatically started by signals identified in Section 6.1.4.3. The two motor driven pumps take suction from a common header and feed all three steam generators through lines which are cross-connected in the pump room as shown in Figure 1. There is a normally closed motor operated valve in each line to the steam generators. The AFWS discharge lines from the motor driven pumps connect to the main feed lines inside containment.

The turbine driven pump takes its source of water from the CST common header and feeds into the main feed system through three normally closed motor-operated valves. The auxiliary feedwater lines from the turbine driven pump train connect to the feedwater regulating valve bypass lines for each individual steam generator outside containment.

The system was evaluated for high energy line breaks in the main steam, main feed lines and the AFWS itself. For the main feed and steam line breaks, at least one train of the AFWS will be able to feed at least one steam generator which is sufficient to safely shut down the plant. Remote manual action would be required to isolate the affected steam generator and AFWS line feeding that generator. For the high energy line break in the Auxiliary Feedwater System, the worst break is in the motor driven pump trains' cross connection line. In this case, since the pumps and associated motor operated valves are in the same room, these trains could be shorted out if the cross connection line is not isolated in time. The steam driven train is, however, still available to shut down the plant, provided the pump does not fail. If the pump fails, auxiliary feedwater flow would be lost, but main feedwater could still be used to supply water to the steam generators.

Sources of Water

There are three sources of water for the auxiliary feedwater system. The primary source is from a 200,000 gallon seismic Category I condensate storage tank (CST), of which 35,000 gallons are dedicated to the auxiliary feedwater system. This will last a minimum of two hours. The CST is not protected against tornado missiles. All valves from the tank to the AFWS are normally open, and are local manually operated valves.

The secondary source of water as well as long term cooling is the seismic Category I service water system and the ultimate heat sink. The piping for this system is buried or in the auxiliary building so it is protected against tornado missiles, however, the pumphouse which contains the service water pumps is not protected against tornado missiles. The valves connecting this system to the AFWS are locked closed manual valves. Thus, it would take time to open these valves. There is, however, sufficient time to open these valves before the condensate storage tank is depleted or the steam generators boil dry.

The back-up source of water is the non-seismic deep well system which has a capacity of 600 gpm. The valves that connect this system with the AFWS are manually locked closed valves.

6.1.2 Components - Design and Classification

Component	Environmental Qualification	Design Classification	Seismic Category
Motor Driven Pump	Ambient	ASME VIII	I
Turbine Driven Pump	"	ASME VIII	I
Piping	"	B31.1	I
Valves/Actuators	"	B31.1	I
Control & Actuation System	"	-	I
Indication	"	-	N.S.
Condensate Storage	"	-	I
Service Water System	"	-	I
Deep Well System	"	-	N.S.
Main Steam Lines to Turbine Driven Pump (connects upstream MSIV)	"	-	I
Main Feed Lines from Main Feed Block Valves to Steam Generators	"	-	I

N.S - Non Seismic Category I

6.1.3 Power Sources

Each motor driven pump is supplied power from its respective emergency bus which receives power from normal station transformers or separate diesel generators (DG). The three motor operated discharge valves (MOV) in the motor driven pump trains are powered from the emergency buses. The valve for steam generator A is powered from emergency bus E1 or E2. The valve for steam generator (SG) B is powered from bus E2 and the valve for SG C is from bus E1. The MOV's are normally closed and fail-as-is. The instrumentation for these trains is taken off the station batteries.

The turbine driven pump is supplied steam from all three steam generators. The steam is taken off upstream of the MSIV and passes through a motor operated valve, a check valve and goes into a common header which feeds the turbine. The motor operated valves take their power from the emergency buses. The valves from SG B&C are connected to bus E1, and the valve from SG A is connected to bus E2. The motor operated valves in the turbine pump discharge lines to the steam generator also take their power from the emergency buses with valves for SG A&C from bus E2 and SG B from bus E1. The above MOV's are normally closed and fail-as-is.

The system does not meet NRC's current power source diversity position with respect to the turbine driven pump train valves although manual action can be taken at the valves. (See recommendation GS-5)

In addition, cooling to the lube oil coolers to the turbine driven pump is from the service water system, which takes its power from the emergency busses. Upon station blackout (loss of all AC), cooling is lost to the turbine which could result in a possible shaft seizure or wiped bearings in the turbine within a short time (approximately 10 to 20 minutes), thus resulting in the loss of all AFW flow. However, the lube oil cooling water piping and valves are arranged so that the lube oil cooler can be cooled by AFW pump flow; but the valve alignment must be changed (See Recommendation GS-5).

6.1.4 Instrumentation and Controls

6.1.4.1 Controls

The following AFWS manual controls are available in the control room:

1. Motor Driven Pump Start-Stop
2. Steam Inlet Line to Turbine - Motor Operated Valves Open-Close
3. AFW Discharge Line Motor Operated Valves - Open-Close

All other valves as well as the above can be controlled at the local stations. Steam generator level is controlled manually at the motor operated discharge valves locally or in the control room by starting and stopping the pumps.

.1.4.2 Information available to Operator

The following information is available to the operator in the control room:

1. Motor Driven Pumps Start-Stop
2. Motor Operated Valves (All) Opened-Closed
3. Motor and Turbine Driven Pumps Discharge Pressure
4. Steam Generator Level
5. Steam Generator and Steam Header Pressure

This information is also available at the local control stations.

The following alarms are located in the control room:

1. Condensate Storage Tank Low Level
2. Equipment on Local Control
3. Steam Generator(s) Low Low Level Alarms
4. Steam Generator High Level Alarm
5. Low AFW Pump Discharge Pressure Alarm and Trip
6. Loss of Lube Oil

X.6.1.4.3 Initiation Signals for Automatic Operation

The following signals initiate automatic operation of the AFWS:

1. Low-Low Level on Steam Generator
    a. 2 out of 3 on one steam generator initiates the motor driven pump trains.
    b. 2 out of 3 on two steam generators initiates the turbine pump train.
2. Loss of Both Main Feed Water Pumps starts Motor Driven Pump Trains.
3. Loss of Offsite Power starts motor driven pump trains.

4. Safety Injection Signal starts motor driven pump trains.

5. Loss of Voltage (<70%) on buses 1 and 4 starts the steam driven pump train. Steam inlet valves (MOV) are operated from buses E1 and E2 and therefore will open.

1.5 Testing

The system is tested on a monthly bases with some exceptions. The pumps are run in recirculating mode monthly and the motor operated discharge valves are stroked monthly with the pumps off. All other valves in the system including the AFW pump turbine steam inlet valves are tested quarterly; however one steam inlet valve is operated monthly in conjunction with the monthly tests of the turbine driven pump. All valves are checked at the end of the tests for correct positioning both in the control room and locally. The piping in the system is hydrostatically tested every 10 years. The system is tested as a whole during refueling cycle as part of the ECCS actuation test. The only pieces of equipment not tested on a periodic bases are the locked closed valves to the service water and deep well systems. The periodic test requirements are as follows:

AUXILIARY FEEDWATER SYSTEM

Applicability

Applies to periodic testing requirements of the turbine-driven and motor-driven auxiliary feedwater pumps.

Objective

To verify the operability of the auxiliary feedwater system and its ability to respond properly when required.

Specification

4.8.1 Each motor driven auxiliary feedwater pump will be started at intervals not to exceed one month, run for 15 minutes, and determined that it is operable.

4.8.2 The steam turbine driven auxiliary feedwater pump by using motor operated steam admission valves will be started at intervals not to exceed one month, run for 15 minutes, and determined that it is operable when the reactor coolant system is above the cold shutdown condition. When periods of reactor cold shutdown extend this interval beyond one month, the test shall be performed immediately following reactor heatup.

4.8.3 The auxiliary feedwater pump discharge valves will be tested by operator action at intervals not greater than one month.

4.8.4 These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

6.1.5 Technical Specifications

The following technical specifications apply to H. B. Robinson. The salient features are that one pump could be out of service for an indefinite period of time with no limiting condition for operation on the plant, that the instrumentation for the system could be out for all trains without limiting condition for operation.

SECONDARY STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of turbine cycle.

Objective

To define conditions of the turbine cycle steam-relieving capacity. Auxiliary Feedwater System and Service Water System operation is necessary to ensure the capability to remove decay heat from the core.

Specification

3.4.1 The reactor coolant shall not be heated above 350°F unless the following conditions are met:

a. A minimum turbine cycle steam relieving capability of twelve (12) main steam safety valves operable.

b. Two of the three auxiliary feedwater pumps must be operable.

c. A minimum of 35,000 gallons of water in the condensate storage tank and an unlimited water supply from the lake via either leg of the plant Service Water System.

d. Essential features including system piping and valves directly associated with the above components are operable.

e. The main steam stop valves are operable and capable of closing in five seconds or less.

3.4.2 The specific activity of the secondary coolant system shall be ≤ 0.10 μCi/gram DOSE EQUIVALENT I-131 under all modes of operation from cold shutdown through power operation. When the specific activity of the secondary coolant system is >0.10 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours.

The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.1-2.

3.4.3 If, during power operations, any of the specifications in 3.4.1 above cannot be met within 24 hours, the operator shall initiate procedures to put the plant in the hot shutdown condition. If any of these specifications cannot be met within 48 hours, the operator shall cool the reactor below 350°F using normal procedures.

2 Reliability Evaluation

2.1 Dominant Failure Modes

The system was analyzed for three cases:

(a) loss of feedwater with offsite power available;

(b) loss of feedwater with onsite AC power available;

(c) loss of feedwater with only DC power available.

The dominant failure modes for each case are summarized below.

X 6.2.1.1 LOFW with Offsite Power available

The dominant failure modes are as follows:

(1) loss of condensate storage tank supply due to failure of valves in the supply line plus failure to manually actuate backup service water supply by locally opening closed valves.

(2) one train out indefinitely for maintenance plus hardware and maintenance outages in other two trains.

X.6.2.1.2 LOFW with Onsite AC power available

The system was analyzed assuming loss of offsite power, considering the possible loss of one of the diesel generators. The dominant failure modes for this case are similar to those discussed in the previous case.

X.6.2.1.3 LOFW with Only DC Power Available

The system will fail in the long-term due to reliance of turbine lube oil cooling on AC power without operator action to realign cooling water valves (See Section 6.1.3). In the short-term (≤45 minutes), unavailability is dominated by maintenance and hardware failures of the turbine driven pump train and failure to manually open the steam and water MOVs which do not open without AC power.

(b) loss of feedwater with onsite AC power available;

(c) loss of feedwater with only DC power available.

The dominant failure modes for each case are summarized below.

X 6.2.1.1 LOFW with Offsite Power available

The dominant failure modes are as follows:

(1) loss of condensate storage tank supply due to failure of valves in the supply line plus failure to manually actuate backup service water supply by locally opening closed valves.

(2) one train out indefinitely for maintenance plus hardware and maintenance outages in other two trains.

X.6.2.1.2 LOFW with Onsite AC power available

The system was analyzed assuming loss of offsite power, considering the possible loss of one of the diesel generators. The dominant failure modes for this case are similar to those discussed in the previous case.

X.6.2.1.3 LOFW with Only DC Power Available

The system will fail in the long-term due to reliance of turbine lube oil cooling on AC power without operator action to realign cooling water valves (See Section 6.1.3). In the short-term ($\leq$45 minutes), unavailability is dominated by maintenance and hardware failures of the turbine driven pump train and failure to manually open the steam and water MOVs which do not open without AC power.

X.6.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumention can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

3. Recommendation GS-4 - Emergency procedures for transferring to alternat sources of AFW supply should be available to the plant operators. Thes procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources shoul take place. The following cases should be covered by the procedures:

- The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

- The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

4. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current

power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed., (See Recommendation GL-3 for the longer-term resolution of this concern.)

5. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system shoul be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circu should be upgraded to meet safety-grade requirements as indicated in

Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to intiate the AFW system from the control room.

7. The licensee should propose modifications to the Technical Specifications to provide for periodic testing of the normally locked closed service water and deep well manual valves.

8. The licensee should propose modifications to the Technical Specifications to provide for monthly testing of all steam admission valves to the turbine pump.

6.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for 1 hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   > "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for

operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.6.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

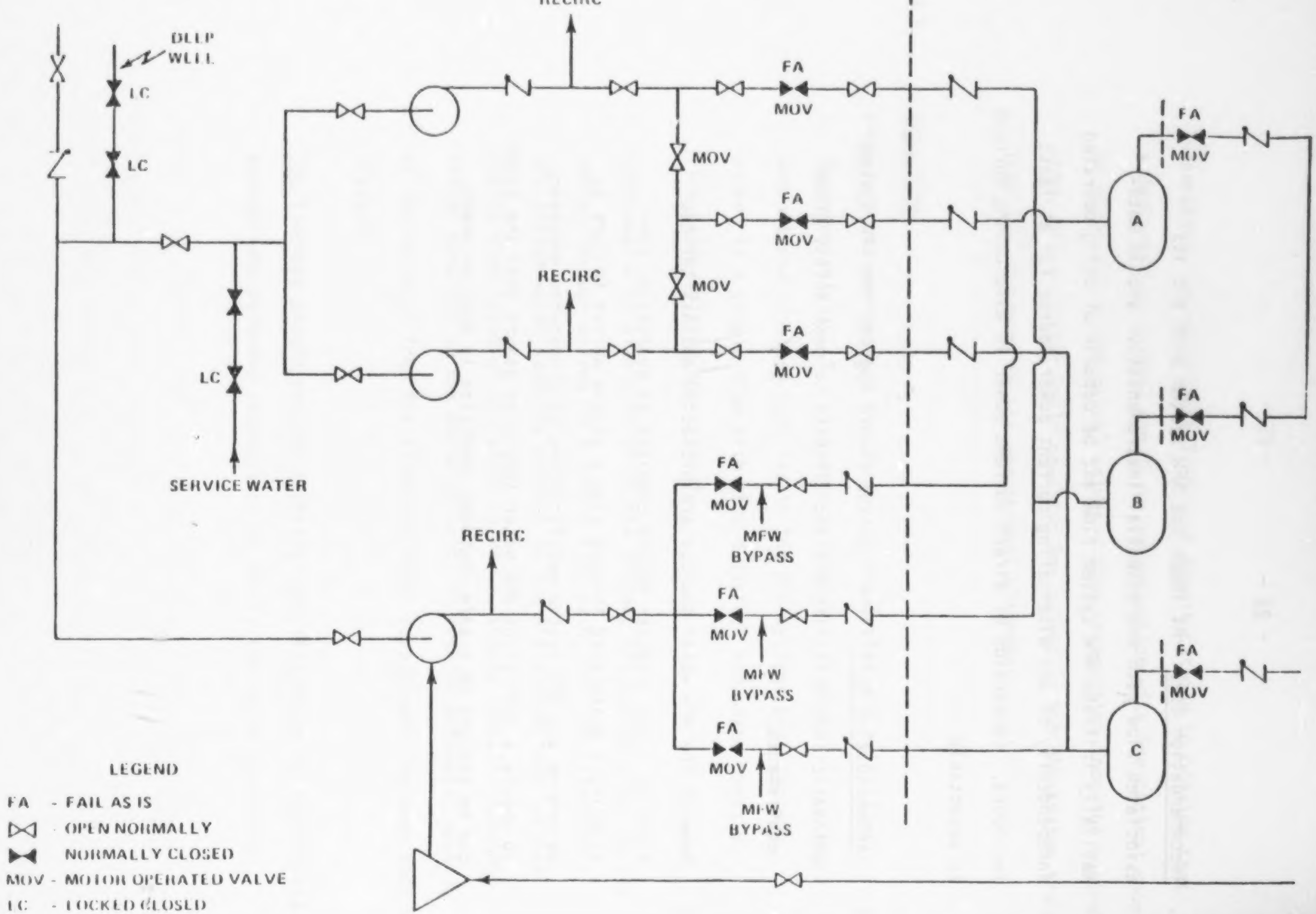
   Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

   The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

3. Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety grade requirements.

4. None of the AFW water sources are protected against tornado missiles. The licensee should complete an evaluation considering a postulated tornado plus a single active failure to determine any AFW system modifications or procedures necessary to assure a sufficient AFW water supply or assure that the plant can be brought to a safe shutdown condition in such an event.


RECIRC
DEEP WELL
LC
LC
FA
MOV
MOV
FA
MOV
RECIRC
MOV
FA
MOV
LC
SERVICE WATER
FA
MOV
MFW BYPASS
RECIRC
FA
MOV
MFW BYPASS
FA
MOV
MFW BYPASS
A
B
C
FA
MOV
FA
MOV
FA
MOV
LEGEND
FA - FAIL AS IS
⋈ OPEN NORMALLY
⧓ NORMALLY CLOSED
MOV - MOTOR OPERATED VALVE
LC - LOCKED CLOSED

Auxiliary Feedwater System
H.B. Robinson

9. INDIAN POINT 2 AUXILIARY FEEDWATER SYSTEM (AFWS)

ENCLOSURE 1

X.7 (W) INDIAN POINT 2 & 3
AUXILIARY FEEDWATER SYSTEM (AFWS)

.7.1 System Description

.7.1.1 Configuration - Overall Design

A simplified drawing of the AFW systems for Units 2 and 3 are shown in Figures 1 and 2 respectively. The system is basically the same for both units, although there are differences in the actuation system and diesel generators. The system consists of one turbine-driven pump (capacity 800 gpm at 1350 psia) and two motor-driven pumps, each with a capacity of 400 gpm at 1350 psia. A flow of 200 gpm to each of two out of four steam generators (SG) is required for safe shutdown. Indian Point Unit 2 and 3 steam generators would boil dry in approximately 35 minutes and 24 minutes respectively without any feedwater flow, assuming a reactor trip.

The AFW water supply consists of a primary source, a secondary source, and a long-range source. The primary source is one seismic Category I condensate storage tank with a total capacity of 600,000 gallons. Of this total volume, 360,000 gal. is dedicated for AFWS use. When the water level in the condensate storage tank reaches the 360,000 gal. low value, a valve automatically closes isolating the condensate storage tank outlet from all other systems. The secondary water source is a 1.5 million gal. city water storage tank which is shared between Units 2 & 3. This backup water supply can be manually initiated from the control room. The long range water source is the city water supply. Each motor-driven AFW pump supplies water to two steam generators. The turbine-driven pump is headered to supply all four steam generators. Motive steam to the turbine-driven pu is from two steam generators; the piping configuration is such that either one or both of these steam generators can provide steam to the turbine-driven pump. The AFW system is automatically actuated, but the operator has to control flow rate to the steam generators remote-manuall

X.7.1.2 Components - Design Classification

The components of the AFWS for IP-2 and IP-3 are classified seismic Class I. The motor-driven pumps and AFW system instrumentation and controls are supplied from Class 1E power sources, except for IP-2 flow control valves (see Section 7.1.4.1 below).

X.7.1.3 Power Sources

In IP-2 and IP-3, the motor-driven pumps receive power from independent AC emergency buses. Pneumatic-operated valves in the steam inlet line to the turbine-driven pump and the AFW flow paths receive power as noted in Figures 1 and 2 and as discussed below.

X.7.1.4 Instrumentation and Controls

X.7.1.4.1 Controls

The AFWS is automatically initiated. Flow control to the four steam generators is through eight air-operated valves located on the discharge side of the pumps, which are normally 35% open. After actuatio of the FW pumps, level in the SGs is maintained manually from the control room by positioning the flow control valves. Each valve can be positioned from the control room via electric/air converters. Each motor-driven auxiliary FW pump has discharge flow paths to two steam generators, each provided with a valve position controller. The turbine-dirven pump has discharge flow paths, each provided with a valve position controller. Air to these valves is from a common header which is supplied by independent air compressors powered from separate emergency diesel generators. The air supply to the valves is backed up by an emergency high pressure nitrogen (bottle) system.

In IP-3, the valve position controllers associated with one motor-driven pump (31) and the turbine-driven pump receive power from independent safety grade instrument buses with backup battery inverters. The controllers associated with the remaining motor-driven pump currently receive power from a safety grade instrument bus. However, the licensee has indicated that a design modification is in progress to supply this bus with a battery inverter system.

In IP-2, all the valve position controllers receive power from the same non-safety grade bus, but fail open on loss of power. (See short term recommendation #7.)

In addition to remote control from the control room, all of the AFW pumps and regulating valves can be operated locally in the auxiliary feedwater building. All regulating valves are equipped with manual operators and equalizing valves for the control air to take the pnuematic operators out of service. With the local steam generator level indication noted in X.7.1.4.2. the level in each of the steam generators can be maintained and controlled from the auxiliary feedwater building without any assistance from the control room.

X.7.1.4.2 Information Available to Operator

System information available to the operator in the control room to assess the performance of the auxiliary feedwater system is as follows:

- Pump on-off-auto trip lights
- Aux feedwater flow path control valves position indication
- Primary source water level indication and alarm
- IP-2 and 3 secondary source high and low water level alarm (alarms located only in IP-2 control room.)
- Aux FW flow indication to each steam senerator
- Steam generator levels

X.7.1.4.3 Initiating Signals for Automatic Operation

IP-3

The auxiliary feedwater pumps are automatically started on receipt of any of the following signals:

Steam Driven Feedwater Pump

1) 2/3 low-low water level in any 2/4 SGs

2) Loss of offsite power concurrent with a main turbine-generator trip

Motor-Driven Feedwater Pumps

1) 2/3 Low-Low Water Level in any one steam generator
2) Loss of either main feed pump
3) Safety injection trip signal
4) Loss of offsite power concurrent with a main turbine-generator trip

IP-2

Steam Driven Feedwater Pump

1) 2/3 Low-Low Water Level in any 2/4 SGs
2) Loss of offsite power concurrent with a main turbine-generator trip.

Motor-Driven Feedwater Pumps

1) 2/3 Low-Low Water Level in any one steam generator
2) Loss of either main feed pump
3) Safety injection trip signal
4) Loss of offsite power concurrent with a main turbine-generator trip

Main steam or main feedwater line break isolation is accomplished automatically in IP-2 and IP-3.

The design of the AFWS does not have the capability to automatically terminate feedwater flow to a depressurized steam generator and provide flow to the intact steam generator. This is accomplished by the operator.

X.7.1.5 Testing

The AFWS is tested periodically in accordance with the following Technical Specification requirements:

Indian Point 2 Specification - Testing Requirements

1.a Each motor-driven auxiliary feedwater pump will be started at intervals not greater than every month with full flow established to the steam generators once every refueling.

b The steam turbine driven auxiliary feedwater pump will be started at intervals not greater than six months with full flow established to the steam generators once every refueling.

c The auxiliary feedwater pumps discharge valves will be tested by operator action at intervals not greater than six months.

2. These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

Indian Point 3 - Specification - Testing Requirements

1.a Each auxiliary feedwater pump will be started manually from the control room at monthly intervals with full flow established to the steam generators once every refueling.

b The auxiliary feedwater pumps discharge valves will be tested by operator action at intervals not greater than six months.

c Backup supply valves from the city water system will be tested once every refueling.

2. Acceptance levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, and operate for at least fifteen minutes.

X.10.1.6 **Technical Specification**

The limiting conditions of operation for Indian Point 2 and 3 AFWS are contained in the following Technical Specifications:

Indian Point 2 - Specification

A. The reactor shall not be heated above 350°F unless the following condition are met:

(1) A minimum ASME code approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing).

(2) Two of the three auxiliary feedwater pumps must be operable.

(3) A minimum of 360,000 gallons of water in the condensate storage tanks and a backup supply from the city water supply.

(4) System piping and valves directly associated with the above components operable.

(5) The main steam stop valves are operable and capable of closing in five seconds or less.

(6) The total iodine activity of I-131 and I-133 on the secondary side of the steam generator shall be less than or equal to 0.15 μCi/cc.

B. If during power operations any of the conditions of 3.4.A above can not be met within 48 hours the operator shall start to shutdown and cool the reactor below 350°F using normal operating procedures.

Indian Point 3 - Specification

A. The reactor shall not be heated above 350°F unless the following condition are met:

(1) A minimum ASME code approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing).

(2) Two of the three auxiliary feedwater pumps must be operable.

(3) A minimum of 360,000 gallons of water in the condensate storage tank.

(4) System piping and valves directly associated with the above components operable.

(5) The main steam stop valves are operable and capable of closing in five seconds or less.

(6) Two steam generators capable of performing their heat transfer function.

(7) City water system piping and valves directly associated with providing backup supply to the auxiliary feedwater pumps are operable.

B. If during power operations any of the conditions of 3.4.A above can not be met within 48 hours the operator shall start to shutdown and cool the reactor below 350°F using normal operating procedures.

X.7.2 Reliability Evaluation

X.7.2.1 Dominant Failure Modes

Loss of MFW with offsite power available

The dominant failure mode for this transient is the failure to provide sufficient water to the suction of the AFWS pumps. There are two manual valves in the supply line from the condensate storage tank; the inadvertent closure of either of these valves cuts off this water supply. In the event of an AFWS demand, operator action would be required to either open the closed valve (locally) or to manually open the valves in the supply line from the alternate water sources (city water) before pump damage occurs. Thus the dominant failure mode is the human error of inadvertently closing a valve in the CST supply line, coupled with the failure of the operator to manually reopen the closed valve or open the valves from the backup water supply.

A second important failure mode was also noted in this evaluation. The Indian Point Technical Specifications and LCOs require only that two of the three AFWS be operable, thus allowing the possibility that one train could be out of service indefinitely. This, in effect, reduces a three train system to a two train system, and thus reduces the predicted AFWS reliability to some degree. Revision of the Technical Specifications/LCOs to the present requirements (in the standard Technical Specifications) would make this failure mode much less significant.

Loss of MFW with only onsite AC power available

Because the dominant failure modes discussed above are not dependent on the source of AC power (onsite or offsite), these modes are also dominant for this transient event.

Loss of MFW with only DC power available

In this transient, loss of both offsite and onsite AC power is postulated to occur, so that the AFWS is reduced to only the steam-driven pump train. Thus failures in this train alone would be sufficient to fail AFWS, for this transient. The dominant failure mode for this case is that the train is out of service for maintenance, for the reason that current Technical Specifications and LCOs specify no time limit that the train could be out of service. Thus the revision of the Technical Specifications and LCOs mentioned for the above cases also would be of significant benefit for this case.

X.7.2.2 Principal Dependencies

The principal dependency found in this analysis is, as discussed above, the manual valves located in the feedwater supply line common to all AFWS pumps and the possible unlimited outage of one pump.

X.7.3 Recommendations for this Plant*

The short-term recommendations (both generic, denoted by GS, and plant-specifi identified in this section represent actions to improve AFW system realiability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL, and plant-specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

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* Recommendations apply to IP-2 and 3 unless otherwise stated.

.7.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

3. Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

   The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

4. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures shouls include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

5. Recommendation GS-6 - The licensee should confirm flow path available ability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

- Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals ans associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation sginals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.
   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.
   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
   - Testability of the initiation signals and circuits shall be a feature of the design.

- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and curcuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

Indian Point 2

7. Recommendation - The pneumatic-operated valves in the steam supply line to the turbine-driven AFW pump, and all of the pneumatic-operated AFW flow control valves derive their power from the same non-safety grade bus. Although these valves are designed to fail open upon the loss of air or power, thereby assuring auxiliary feedwater flow to the steam generators upon such losses, it cannot be concluded that all failures will result in opening the valves. The consequences of voltage degradation should be analyzed as well as other failures (e.g., restricted air flow)

to assure tha' such events would not incapacitate the auxiliary feedwater system the licensee should establish suitable emergency procedures to assure AFWS function for such events.

X.7.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accompolished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria

should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operatin should propose Technical Specification to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.7.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

   Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s) should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

   The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

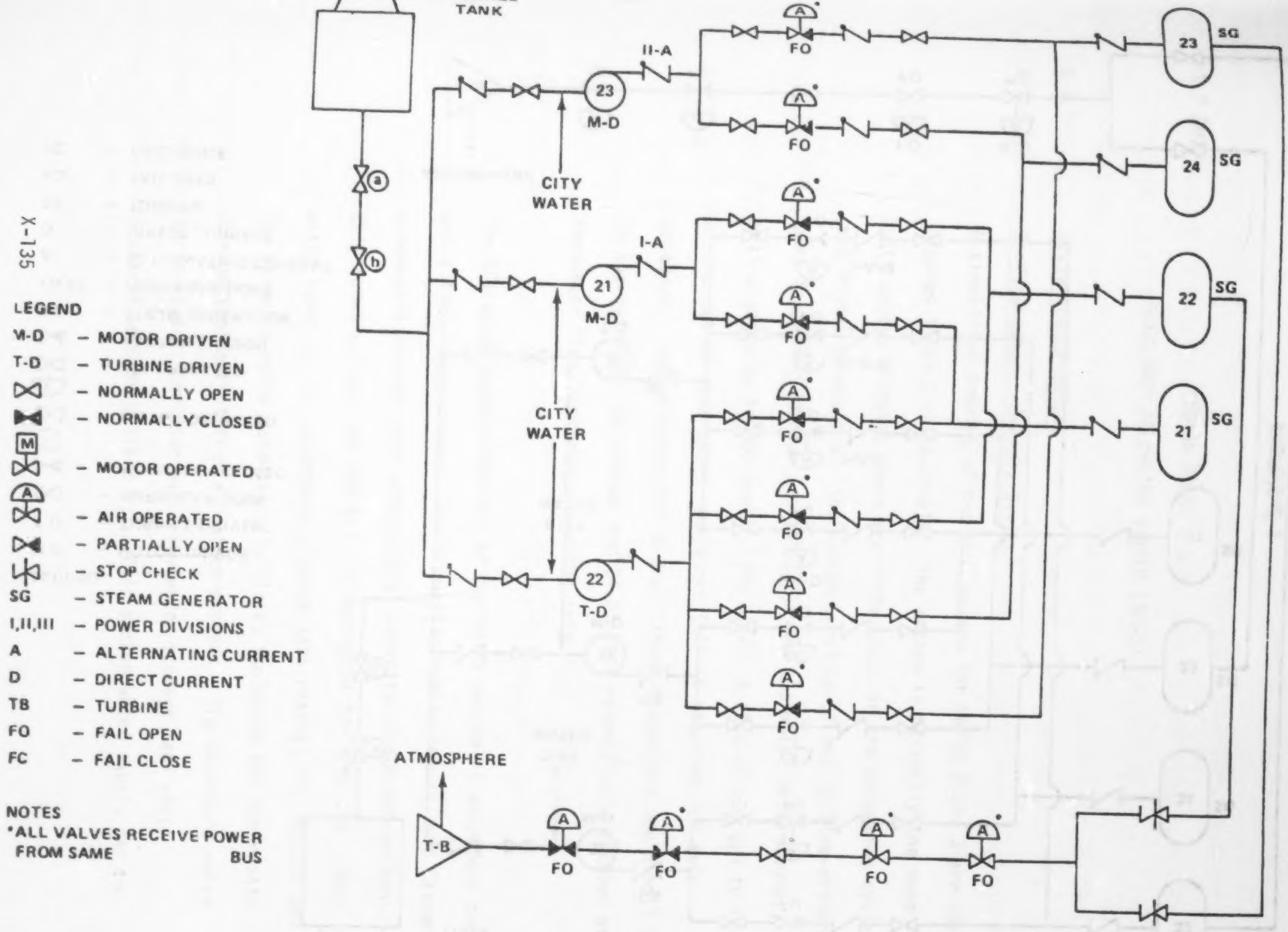
3. Recommendation - The two motor-driven pumps and the turbine driven pump are located in the same room. The licensee should evaluate the capability of the design to withstand a) environmental conditions (steam, flooding, pipe whip and jet impingement) resulting from a pipe break, b) internally generated missiles.

The licensee should evaluate the postulated pipe breaks stated above and (1) determine any AFW system design changes or procedures necessary to detect and isolate the break and direct the required feedwater flow to the steam generator(s) before they boild dry or (2) describe how the plant can be brought to a safe shutdown condition by use of other systems which would be available following such postulated events.

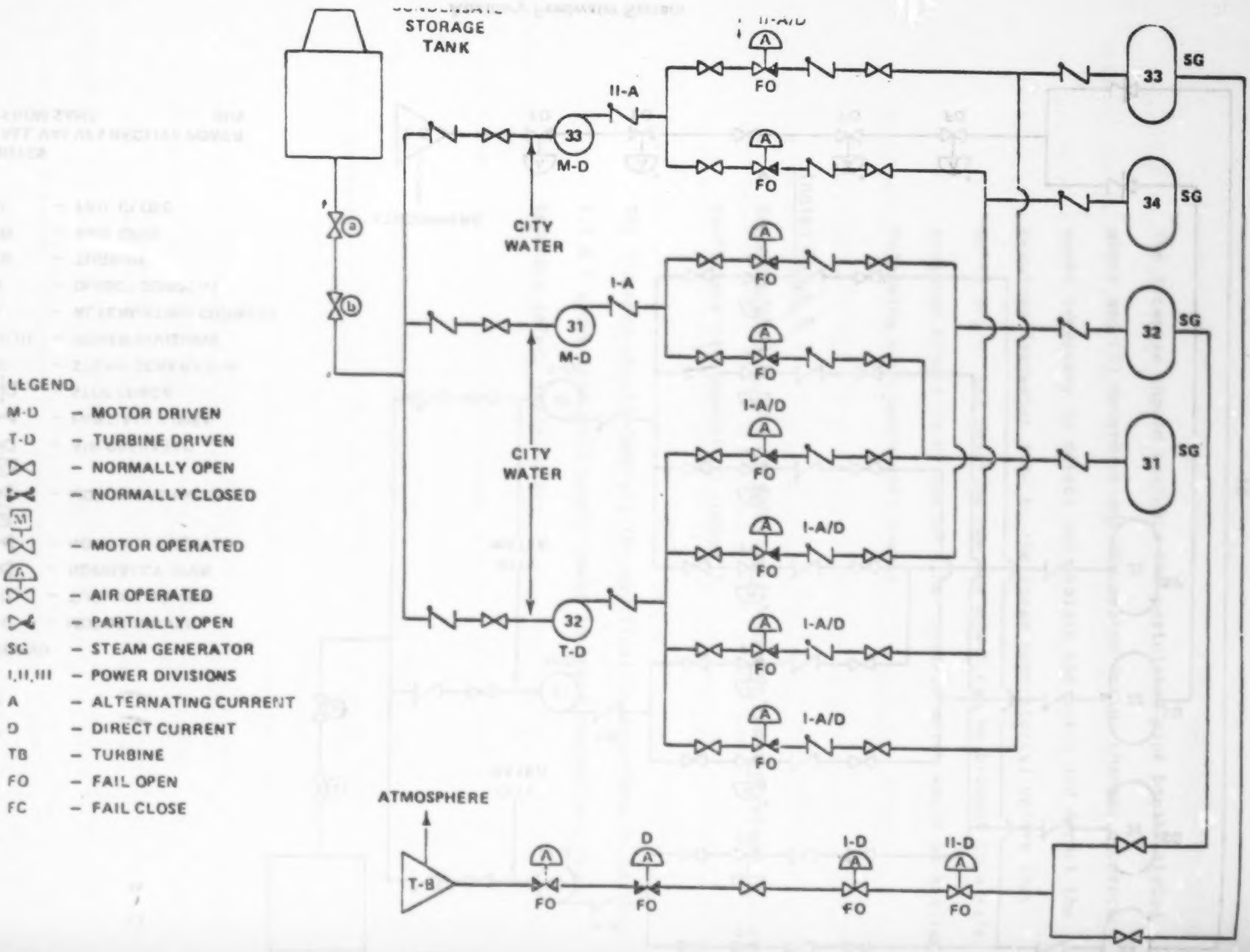
Indian Point 2

4. Recommendation - This is the same concern as that addressed in short term recommendation number 7.

The licensee should complete the modification described in Section 7.1.4.1 above that will supply power to these controllers from separate safety grade buses.


TANK
(a)
(h)
CITY WATER
CITY WATER
23
M-D
21
M-D
22
T-D
II-A
I-A
A
FO
23
24
22
21
SG
SG
SG
SG
ATMOSPHERE
T-B
LEGEND
M-D – MOTOR DRIVEN
T-D – TURBINE DRIVEN
– NORMALLY OPEN
– NORMALLY CLOSED
M – MOTOR OPERATED
A – AIR OPERATED
– PARTIALLY OPEN
– STOP CHECK
SG – STEAM GENERATOR
I,II,III – POWER DIVISIONS
A – ALTERNATING CURRENT
D – DIRECT CURRENT
TB – TURBINE
FO – FAIL OPEN
FC – FAIL CLOSE
NOTES
*ALL VALVES RECEIVE POWER FROM SAME BUS

Auxiliary Feedwater System
Indian Point-2


STORAGE TANK
CITY WATER
CITY WATER
33 M-D
31 M-D
32 T-D
II-A
I-A
II-A/D
I-A/D
FO
SG 33
SG 34
SG 32
SG 31
ATMOSPHERE
T-B
D
I-D
II-D
a
b
LEGEND
M-D — MOTOR DRIVEN
T-D — TURBINE DRIVEN
— NORMALLY OPEN
— NORMALLY CLOSED
M — MOTOR OPERATED
A — AIR OPERATED
— PARTIALLY OPEN
SG — STEAM GENERATOR
I,II,III — POWER DIVISIONS
A — ALTERNATING CURRENT
D — DIRECT CURRENT
TB — TURBINE
FO — FAIL OPEN
FC — FAIL CLOSE

ENCLOSURE 1

X.7 (W) INDIAN POINT 2 & 3 AUXILIARY FEEDWATER SYSTEM (AFWS)

X.7.1 System Description

X.7.1.1 Configuration - Overall Design

A simplified drawing of the AFW systems for Units 2 and 3 are shown in Figures 1 and 2 respectively. The system is basically the same for both units, although there are differences in the actuation system and diesel generators. The system consists of one turbine-driven pump (capacity 800 gpm at 1350 psia) and two motor-driven pumps, each with a capacity of 400 gpm at 1350 psia. A flow of 200 gpm to each of two out of four steam generators (SG) is required for safe shutdown. Indian Point Unit 2 and 3 steam generators would boil dry in approximately 35 minutes and 24 minutes respectively without any feedwater flow, assuming a reactor trip.

The AFW water supply consists of a primary source, a secondary source, and a long-range source. The primary source is one seismic Category I condensate storage tank with a total capacity of 600,000 gallons. Of this total volume, 360,000 gal. is dedicated for AFWS use. When the water level in the condensate storage tank reaches the 360,000 gal. low value, a valve automatically closes isolating the condensate storage tank outlet from all other systems. The secondary water source is a 1.5 million gal. city water storage tank which is shared between Units 2 & 3. This backup water supply can be manually initiated from the control room. The long range water source is the city water supply. Each motor-driven AFW pump supplies water to two steam generators. The turbine-driven pump is headered to supply all four steam generators. Motive steam to the turbine-driven pump is from two steam generators; the piping configuration is such that either one or both of these steam generators can provide steam to the turbine-driven pump. The AFW system is automatically actuated, but the operator has to control flow rate to the steam generators remote-manually.

X.7.1.2 Components - Design Classification

The components of the AFWS for IP-2 and IP-3 are classified seismic Class I. The motor-driven pumps and AFW system instrumentation and controls are supplied from Class 1E power sources, except for IP-2 flow control valves (see Section 7.1.4.1 below).

X.7.1.3 Power Sources

In IP-2 and IP-3, the motor-driven pumps receive power from independent AC emergency buses. Pneumatic-operated valves in the steam inlet line to the turbine-driven pump and the AFW flow paths receive power as noted in Figures 1 and 2 and as discussed below.

X.7.1.4 Instrumentation and Controls

X.7.1.4.1 Controls

The AFWS is automatically initiated. Flow control to the four steam generators is through eight air-operated valves located on the discharge side of the pumps, which are normally 35% open. After actuation of the FW pumps. level in the SGs is maintained manually from the control room by positioning the flow control valves. Each valve can be positioned from the control room via electric/air converters. Each motor-driven auxiliary FW pump has discharge flow paths to two steam generators, each provided with a valve position controller. The turbine-dirven pump has discharge flow paths, each provided with a valve position controller. Air to these valves is from a common header which is supplied by independent air compressors powered from separate emergency diesel generators. The air supply to the valves is backed up by an emergency high pressure nitrogen (bottle) system.

In IP-3, the valve position controllers associated with one motor-driven pump (31) and the turbine-driven pump receive power from independent safety grade instrument buses with backup battery inverters. The controllers associated with the remaining motor-driven pump currently receive power from a safety grade instrument bus. However, the licensee has indicated that a design modification is in progress to supply this bus with a battery inverter system.

In IP-2, all the valve position controllers receive power from the same non-safety grade bus, but fail open on loss of power. (See short term recommendation #7.)

In addition to remote control from the control room, all of the AFW pumps and regulating valves can be operated locally in the auxiliary feedwater building. All regulating valves are equipped with manual operators and equalizing valves for the control air to take the pnuematic operators out of service. With the local steam generator level indication noted in X.7.1.4.2, the level in each of the steam generators can be maintained and controlled from the auxiliary feedwater building without any assistance from the control room.

X.7.1.4.2 Information Available to Operator

System information available to the operator in the control room to assess the performance of the auxiliary feedwater system is as follows:

- Pump on-off-auto trip lights
- Aux feedwater flow path control valves position indication
- Primary source water level indication and alarm
- IP-2 and 3 secondary source high and low water level alarm (alarms located only in IP-2 control room.)
- Aux FW flow indication to each steam senerator
- Steam generator levels

X.7.1.4.3 Initiating Signals for Automatic Operation

IP-3

The auxiliary feedwater pumps are automatically started on receipt of any of the following signals:

Steam Driven Feedwater Pump

1) 2/3 low-low water level in any 2/4 SGs

2) Loss of offsite power concurrent with a main turbine-generator trip

Motor-Driven Feedwater Pumps

1) 2/3 Low-Low Water Level in any one steam generator

2) Loss of either main feed pump

3) Safety injection trip signal

4) Loss of offsite power concurrent with a main turbine-generator trip

IP-2

Steam Driven Feedwater Pump

1) 2/3 Low-Low Water Level in any 2/4 SGs

2) Loss of offsite power concurrent with a main turbine-generator trip.

Motor-Driven Feedwater Pumps

1) 2/3 Low-Low Water Level in any one steam generator

2) Loss of either main feed pump

3) Safety injection trip signal

4) Loss of offsite power concurrent with a main turbine-generator trip

Main steam or main feedwater line break isolation is accomplished automatically in IP-2 and IP-3.

The design of the AFWS does not have the capability to automatically terminate feedwater flow to a depressurized steam generator and provide flow to the intact steam generator. This is accomplished by the operator.

X.7.1.5 Testing

The AFWS is tested periodically in accordance with the following Technical Specification requirements:

Indian Point 2 Specification - Testing Requirements

1.a Each motor-driven auxiliary feedwater pump will be started at intervals not greater than every month with full flow established to the steam generators once every refueling.

b The steam turbine driven auxiliary feedwater pump will be started at intervals not greater than six months with full flow established to the steam generators once every refueling.

c The auxiliary feedwater pumps discharge valves will be tested by operator action at intervals not greater than six months.

2. These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

Indian Point 3 - Specification - Testing Requirements

1.a Each auxiliary feedwater pump will be started manually from the control room at monthly intervals with full flow established to the steam generators once every refueling.

b The auxiliary feedwater pumps discharge valves will be tested by operator action at intervals not greater than six months.

c Backup supply valves from the city water system will be tested once every refueling.

2. Acceptance levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, and operate for at least fifteen minutes.

10.1.6 Technical Specification

The limiting conditions of operation for Indian Point 2 and 3 AFWS are contained in the following Technical Specifications:

Indian Point 2 - Specification

A. The reactor shall not be heated above 350°F unless the following condition are met:

(1) A minimum ASME code approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing).

(2) Two of the three auxiliary feedwater pumps must be operable.

(3) A minimum of 360,000 gallons of water in the condensate storage tanks and a backup supply from the city water supply.

(4) System piping and valves directly associated with the above components operable.

(5) The main steam stop valves are operable and capable of closing in five seconds or less.

(6) The total iodine activity of I-131 and I-133 on the secondary side of the steam generator shall be less than or equal to 0.15 μCi/cc.

B. If during power operations any of the conditions of 3.4.A above can not be met within 48 hours the operator shall start to shutdown and cool the reactor below 350°F using normal operating procedures.

Indian Point 3 - Specification

A. The reactor shall not be heated above 350°F unless the following condition are met:

(1) A minimum ASME code approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing).

(2) Two of the three auxiliary feedwater pumps must be operable.

(3) A minimum of 360,000 gallons of water in the condensate storage tank.

(4) System piping and valves directly associated with the above components operable.

(5) The main steam stop valves are operable and capable of closing in five seconds or less.

(6) Two steam generators capable of performing their heat transfer function.

(7) City water system piping and valves directly associated with providing backup supply to the auxiliary feedwater pumps are operable.

B. If during power operations any of the conditions of 3.4.A above can not be met within 48 hours the operator shall start to shutdown and cool the reactor below 350°F using normal operating procedures.

X.7.2 Reliability Evaluation

X.7.2.1 Dominant Failure Modes

Loss of MFW with offsite power available

The dominant failure mode for this transient is the failure to provide sufficient water to the suction of the AFWS pumps. There are two manual valves in the supply line from the condensate storage tank; the inadvertent closure of either of these valves cuts off this water supply. In the event of an AFWS demand, operator action would be required to either open the closed valve (locally) or to manually open the valves in the supply line from the alternate water sources (city water) before pump damage occurs. Thus the dominant failure mode is the human error of inadvertently closing a valve in the CST supply line, coupled with the failure of the operator to manually reopen the closed valve or open the valves from the backup water supply.

A second important failure mode was also noted in this evaluation. The Indian Point Technical Specifications and LCOs require only that two of the three AFWS be operable, thus allowing the possibility that one train could be out of service indefinitely. This, in effect, reduces a three train system to a two train system, and thus reduces the predicted AFWS reliability to some degree. Revision of the Technical Specifications/LCOs to the present requirements (in the standard Technical Specifications) would make this failure mode much less significant.

Loss of MFW with only onsite AC power available

Because the dominant failure modes discussed above are not dependent on the source of AC power (onsite or offsite), these modes are also dominant for this transient event.

Loss of MFW with only DC power available

In this transient, loss of both offsite and onsite AC power is postulated to occur, so that the AFWS is reduced to only the steam-driven pump train. Thus failures in this train alone would be sufficient to fail AFWS, for this transient. The dominant failure mode for this case is that the train is out of service for maintenance, for the reason that current Technical Specifications and LCOs specify no time limit that the train could be out of service. Thus the revision of the Technical Specifications and LCOs mentioned for the above cases also would be of significant benefit for this case.

X.7.2.2 Principal Dependencies

The principal dependency found in this analysis is, as discussed above, the manual valves located in the feedwater supply line common to all AFWS pumps and the possible unlimited outage of one pump.

X.7.3 Recommendations for this Plant*

The short-term recommendations (both generic, denoted by GS, and plant-specif identified in this section represent actions to improve AFW system realiability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL, and plant-specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

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* Recommendations apply to IP-2 and 3 unless otherwise stated.

.7.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

3. Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

   The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

4. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures shouls include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

5. Recommendation GS-6 - The licensee should confirm flow path available ability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals ans associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation sginals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.
   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
   - Testability of the initiation signals and circuits shall be a feature of the design.

The initiation signals and circuits should be powered from the emergency buses.

- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and curcuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

Indian Point 2

7. Recommendation - The pneumatic-operated valves in the steam supply line to the turbine-driven AFW pump, and all of the pneumatic-operated AFW flow control valves derive their power from the same non-safety grade bus. Although these valves are designed to fail open upon the loss of air or power, thereby assuring auxiliary feedwater flow to the steam generators upon such losses, it cannot be concluded that all failures will result in opening the valves. The consequences of voltage degradation should be analyzed as well as other failures (e.g., restricted air flow)

to assure that such events would not incapacitate the auxiliary feedwater system the licensee should establish suitable emergency procedures to assure AFWS function for such events.

X.7.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accompolished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour Test acceptance criteria

should include demonstrating that the pumps emain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUPEG-0578:

> "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operatin should propose Technical Specification to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

.7.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

   Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s) should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

   The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

3. Recommendation - The two motor-driven pumps and the turbine driven pump are located in the same room. The licensee should evaluate the capability of the design to withstand a) environmental conditions (steam, flooding, pipe whip and jet impingement) resulting from a pipe break, b) internally generated missiles.

The licensee should evaluate the postulated pipe breaks stated above and (1) determine any AFW system design changes or procedures necessary to detect and isolate the break and direct the required feedwater flow to the steam generator(s) before they boild dry or (2) describe how the plant can be brought to a safe shutdown condition by use of other systems which would be available following such postulated events.

Indian Point 2

4. Recommendation - This is the same concern as that addressed in short term recommendation number 7.

The licensee should complete the modification described in Section 7.1.4.1 above that will supply power to these controllers from separate safety grade buses.

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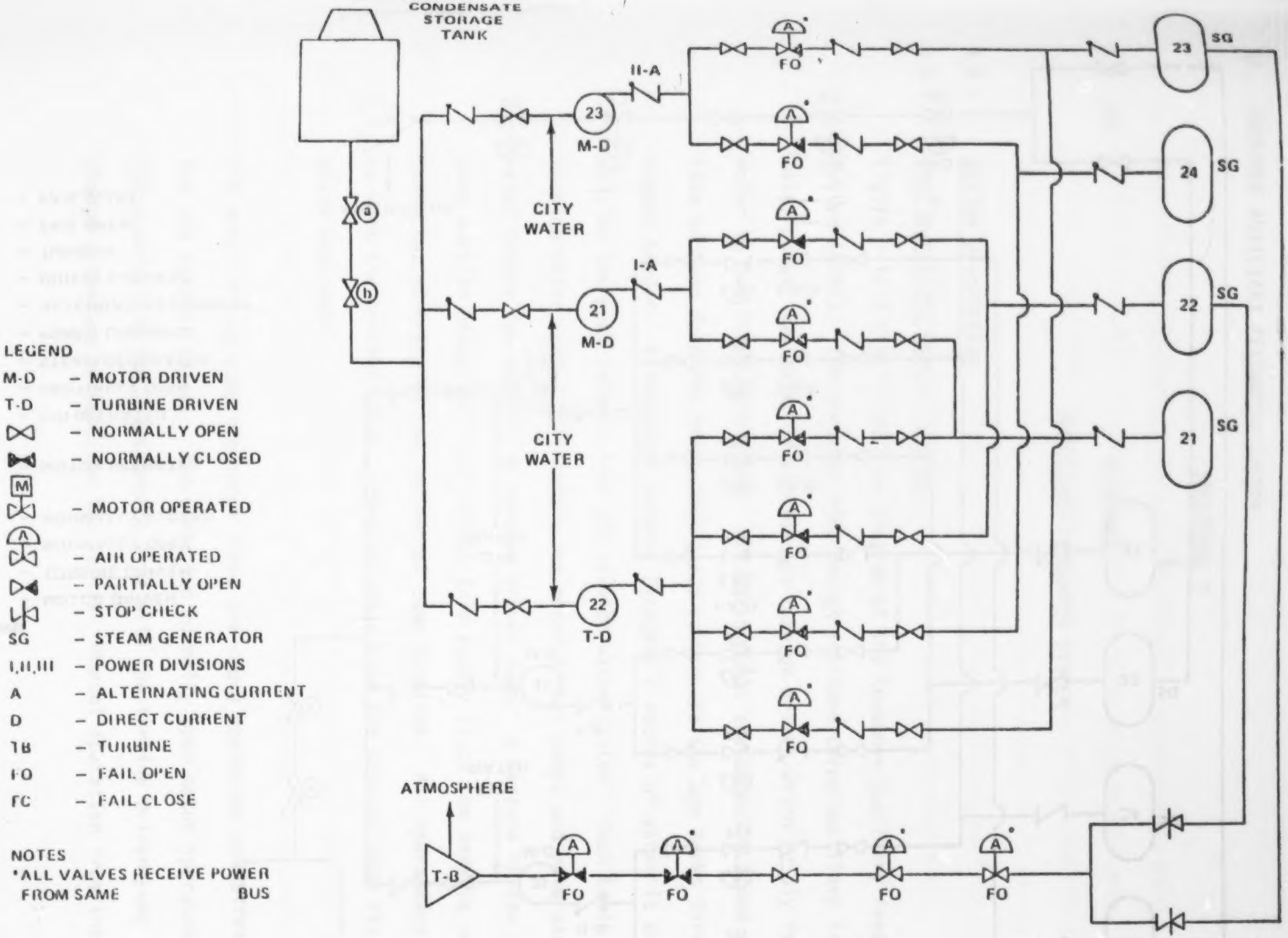
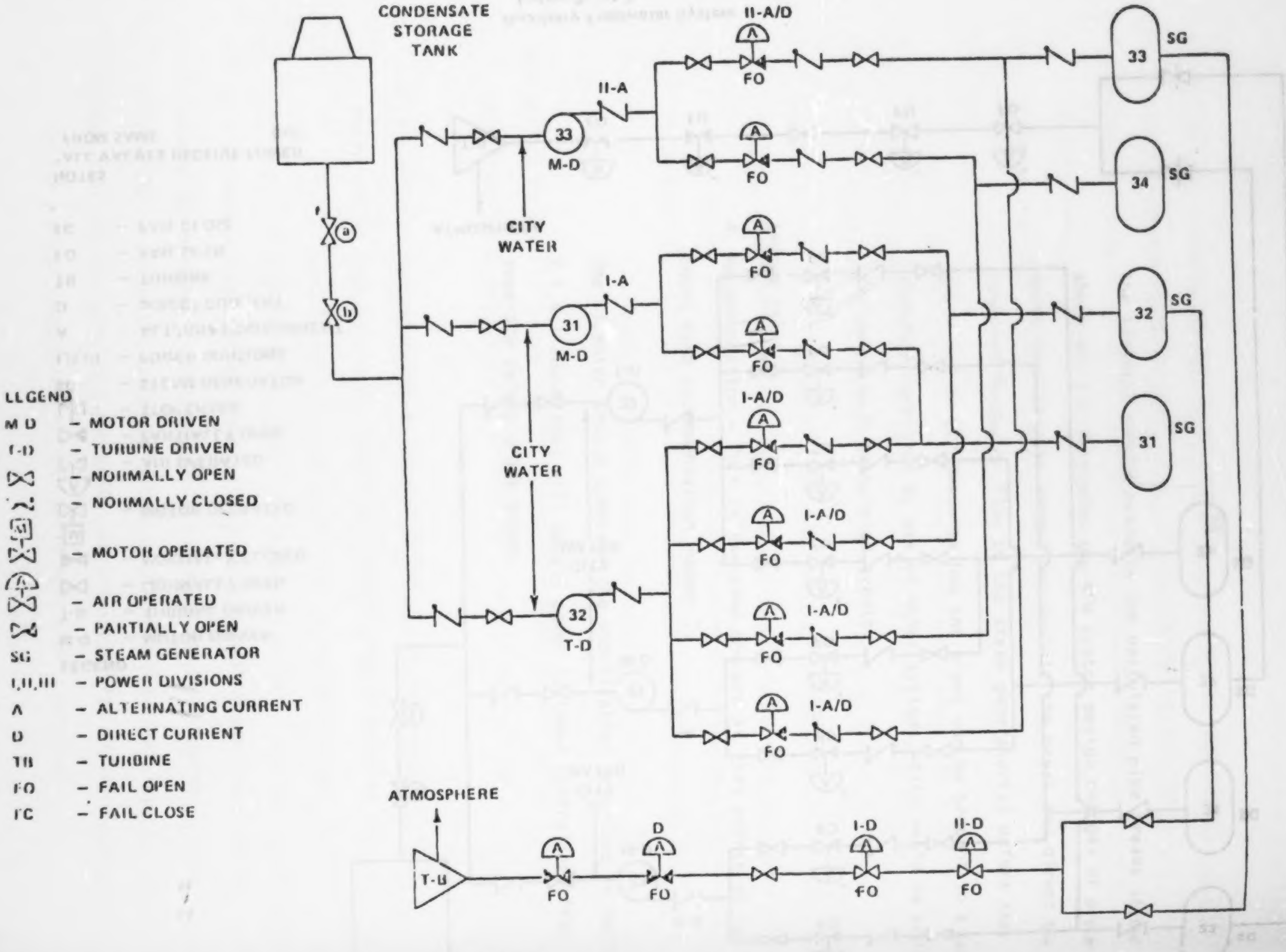

CONDENSATE
STORAGE
TANK
a
b
CITY WATER
CITY WATER
23
M-D
21
M-D
22
T-D
II-A
I-A
A
FO
23
SG
24
SG
22
SG
21
SG
ATMOSPHERE
T-B
LEGEND
M-D – MOTOR DRIVEN
T-D – TURBINE DRIVEN
– NORMALLY OPEN
– NORMALLY CLOSED
M
– MOTOR OPERATED
A
– AIR OPERATED
– PARTIALLY OPEN
– STOP CHECK
SG – STEAM GENERATOR
I,II,III – POWER DIVISIONS
A – ALTERNATING CURRENT
D – DIRECT CURRENT
TB – TURBINE
FO – FAIL OPEN
FC – FAIL CLOSE
NOTES
*ALL VALVES RECEIVE POWER FROM SAME BUS

Auxiliary Feedwater System
Indian Point-2
Figure 1


CONDENSATE STORAGE TANK
a
b
CITY WATER
CITY WATER
33 M-D
31 M-D
32 T-D
II-A
I-A
II-A/D
A
FO
I-A/D
SG 33
SG 34
SG 32
SG 31
ATMOSPHERE
T-B
D
I-D
II-D
LEGEND
M D – MOTOR DRIVEN
T-D – TURBINE DRIVEN
– NORMALLY OPEN
– NORMALLY CLOSED
– MOTOR OPERATED
– AIR OPERATED
– PARTIALLY OPEN
SG – STEAM GENERATOR
I,II,III – POWER DIVISIONS
A – ALTERNATING CURRENT
D – DIRECT CURRENT
TB – TURBINE
FO – FAIL OPEN
FC – FAIL CLOSE

Auxiliary Feedwater System

11. KEWAUNEE AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

X.8 (W) KEWAUNEE AUXILIARY FEEDWATER SYSTEM

X.8.1 System Description

X.8.1.1 Configuration, Overall Design

Figure 1 is a simplified flow diagram of the Kewaunee auxiliary feedwater system (AFWS). The Kewaunee AFWS design includes three auxiliary feedwater pumps, two motor driven and one turbine driven, which supply feedwater to two steam generators. All three pumps are normally lined up to take suction from the non-safety grade condensate storage tanks through a common header. A redundant seismic Category I source of water is available to the pump suction from the service water system. Each train of the service water system will supply one motor driven pump, and both service water trains can supply the turbine driven pump. A failure in the common pump suction header will not affect this supply since the service water is connected directly to the individual pump suctions. All operations to connect the service water system are done from the control room via safety grade equipment.

The motor driven pump discharge lines are cross connected downstream of the AFW control valves. There are two normally open motor operated valves (DC powered) in the crossconnect line with the turbine driven pump discharge connected between the two valves. Manual operation from the control room is available to separate the headers or direct the turbine driven pump discharge to a specific steam generator. This is accomplished by closing one or both of the cross connect isolation valves.

Each motor driven pump (240 gpm) has its own air operated flow control valve located at the pump discharge. These valves are normally open and fail open on loss of air. Each of the three pumps has an individual normally open manual isolation valve at the pump discharge. There is no flow control valve for the turbine driven pump which operates at full flow capacity continuously on demand (240 gpm).

There are no other valves in the flowpath between the pumps and the steam generators other than a check valve at each pump discharge and a check valve in each of the two discharge lines at the main feedwater system connection.

There are two condensate storage tanks (capacity 75,000 gallons each). Technical specifications require at least 75,000 gallons total be available when the reactor is above 350°F or the plant must be cooled down below 350° within 48 hours. One condensate tank is normally lined up to the AFW system suction, with the other tank is used for normal secondary system demands

Each motor pump uses a startup lube oil pump, powered from the same electrical train as that of the pump. The turbine driven start-up lube oil pump is d-c powered such that the turbine driven train is independent of A-C power. Electrical interlocks are provided such that a pump will not start unless its respective lube oil startup pump develops sufficient lube oil pressure. Once up to speed, lube oil can be supplied by shaft driven lube oil pumps independent of the startup system. While running, a loss of lube oil pressure will result in a pump trip.

X.8.1.2 Components, Design Classification

The condensate storage tank is classified non-safety grade and non-seismic Category I. The AFW piping, valves and pumps are classified safety Class I and seismic Category I. All safety systems at Kewaunee, including the reactor coolant system, are classified safety Class 1 and have the same quality assurance requirements. Steam generator level indication, valve position indication and all control equipment are safety grade. The rest of the system instrumentation is non-safety grade.

X.8.1.3 Power Sources

The valves which initiate steam flow to the turbine driven pump are operated by a safety grade d-c power supply. These are the normally open motor operated isolation valves from the steam lines and a normally closed motor operated valve at the turbine inlet. Each motor driven pump receives power from separate emergency A-C buses capable of being supplied by the diesel generators.

The normally open flow control valves from each motor driven pump are air operated and fail open on loss of air. Control power is from the same emergency bus as its respective pump. The cross connect isolation valves are D-C powered motor operated valves supplied by safety related D-C buses.

All instrumentation and controls associated with the auxiliary feedwater systems are powered from onsite electrical systems.

X.8.1.4 Instrumentation and Controls

X.8.1.4.1 Controls

The Kewaunee AFW design has a minimum number of control features because of the small number of valves associated with the system. These control are:

1) Motor Driven Pump Start/Stop Switches
2) Turbine driven Pump Steam inlet valve open/close for start and stop operations
3) Modulation Control of AFW flow control valves for motor driven pumps from full open to full closed
4) Open/Close control of discharge header cross connect isolation valves (each isolation valves will also isolate turbine pump discharge from individual steam generators).
5) Open/Close control of turbine steam isolation valves which isolate turbine from main steam lines.
6) Open/Close control of service water system isolation valves to AFW pump suctions.

All controls are located in the control room. Steam generator level will normally be controlled by modulating the flow control valves on the discharge of the motor driven pumps. The turbine driven pump will deliver full flow when operating and will be secured manually from the control room if flow to generators is more than necessary.

X.8.1.4.2 Information Available to Operator

The following alarms and indications are available to the operator in the control room:

I. Indication

1) System Actuation Light - Actuates when motor is energized with sufficient pump discharge pressure. (Both signals necessary)
2) Discharge Pressure for each pump
3) Flow rate to each steam generator
4) Condensate storage tanks level indication
5) Steam Generator Level
6) Valve position indication for steam valves to turbine, feedwater control valves, crossconnect isolation valves and service water system supply to AFW isolation valves.

II. Alarms (Turbine Pump)

1) Steam inlet valve open to turbine coincident with low lube oil pressure to turbine driven pump
2) Steam inlet valve open coincident with low discharge pressure at turbine driven pump

3) Steam isolation valve (either of two) from main steam system to turbine header not open

4) Turbine throttle valve - not open

5) Discharge valve not open - either of two crossconnect isolation valves

6) Valve Control Power - In pullout position (cannot actuate valve at steam inlet)

III. Alarms (Motor Driven pumps)

1) Breaker closed coincident with low lube oil pressure

2) Breaker closed Low Discharge Pressure

3) Breaker in pull out - Pump cannot be started

4) Breaker open coincident with auxiliary feedwater control valve closed.

X.8.1.4.3 Initiating Signals for Automatic Operation

I. Motor Driven Pumps

1) Safety Injection Signal

2) Loss of Bus Voltage (Loss of Offsite Power - LOOP)

3) Low-Low Level in either steam generator-two out of three detectors

4) Tripping of Both Main feedwater pumps. (Motor Driven Main Feed Pumps - Signal taken from contact on circuit breaker)

II. Turbine Driven Pump

1) Low-Low level in both steam generators-2 out of 3 detectors

2) Loss of Voltage on both 4-KV busses (Reactor Coolant Pump and Main Feedwater Pump supply bus)

All pumps can be manually started from the control room and will automatically supply full flow to steam generators when started.

The main feedwater pump trip automatic start of the motor driven auxiliary feedwater pumps is bypassed during startup by breaker "pull out" switch in control room.

X.8.1.5 Testing

Pump operability is tested once per month, by closing the auxiliary feedwater control valves and manual isolation valve at pump discharge and verifying discharge pressure while recirculating to the condensate storage tank. This same test also verifies valve operability. The service water system isolation valves to the auxiliary feedwater pumps are cycled quarterly to verify operability.

X.8.1.6 Technical Specifications

Limiting conditions for Operation with regard to the auxiliary feedwater system are:

1) The reactor shall not be above 350°F unless the following conditions are met.

   a) Two of three AFW pumps are operable
   b) System piping and valves for 2 pump trains are available
   c) Minimum of 75,000 gallons of water is available in the condensate storage tanks and the service water system is capable of delivering an unlimited supply from Lake Michigan.

2) If, when the reactor is above 350°F, any of the above conditions are not met within 48 hours, the reactor shall be shutdown and cooled to below 350° using normal operating procedures.

X.8.2 Reliability Evaluation

X.8.2.1 Dominant Failure Modes

The dominant failure modes are expressed for three transient situations. Success criterion is the operation of at least one of the three pump trains.

LOFW with Offsite Power Available

The unreliability of the AFWS during this type of transient is dominated by two types of failure combinations. The first involves initiation of the AFW pumps with inadvertent closure, and delayed discovery of the manual valves in the CST supply line combined with human failure to switch to the service water source.

The second failure combination type is based on maintenance outages combined with hardware failure. The Kewaunee technical specification permits unlimited outage of one of the three subsystems and permits outage or test of a second system for up to 48 hours prior to a required shutdown.

LOFW With Loss of Offsite Power but With Onsite AC Power Available

The conditional unreliability of the AFWS during this type of transient is dominated by the same failure mode as in the previous section with a more significant contribution from the triple hardware failure. In this situation failure of one of the two electrical loops can come from partial (one train) failure of onsite power.

LOFW with Loss of All AC, DC Available

The conditional unreliability of the AFWS during this type of transient is dominated by the test and maintenance contribution by the turbine driven pump train. Since only the turbine driven pump train is useable under these conditions, unlimited possible outage of that train makes a high probability for AFWS outage possible.

X.8.2.2 Interdependencies

None noted.

X.8.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.8.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

3. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

- The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
- The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

- Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.
   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
   - Testability of the initiation signals and circuits shall be a feature of the design.
   - The initiation signals and circuits should be powered from the emergency buses.

- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to intiate the AFW system from the control room.

X.8.3.2 Additional Short Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditons (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.8.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

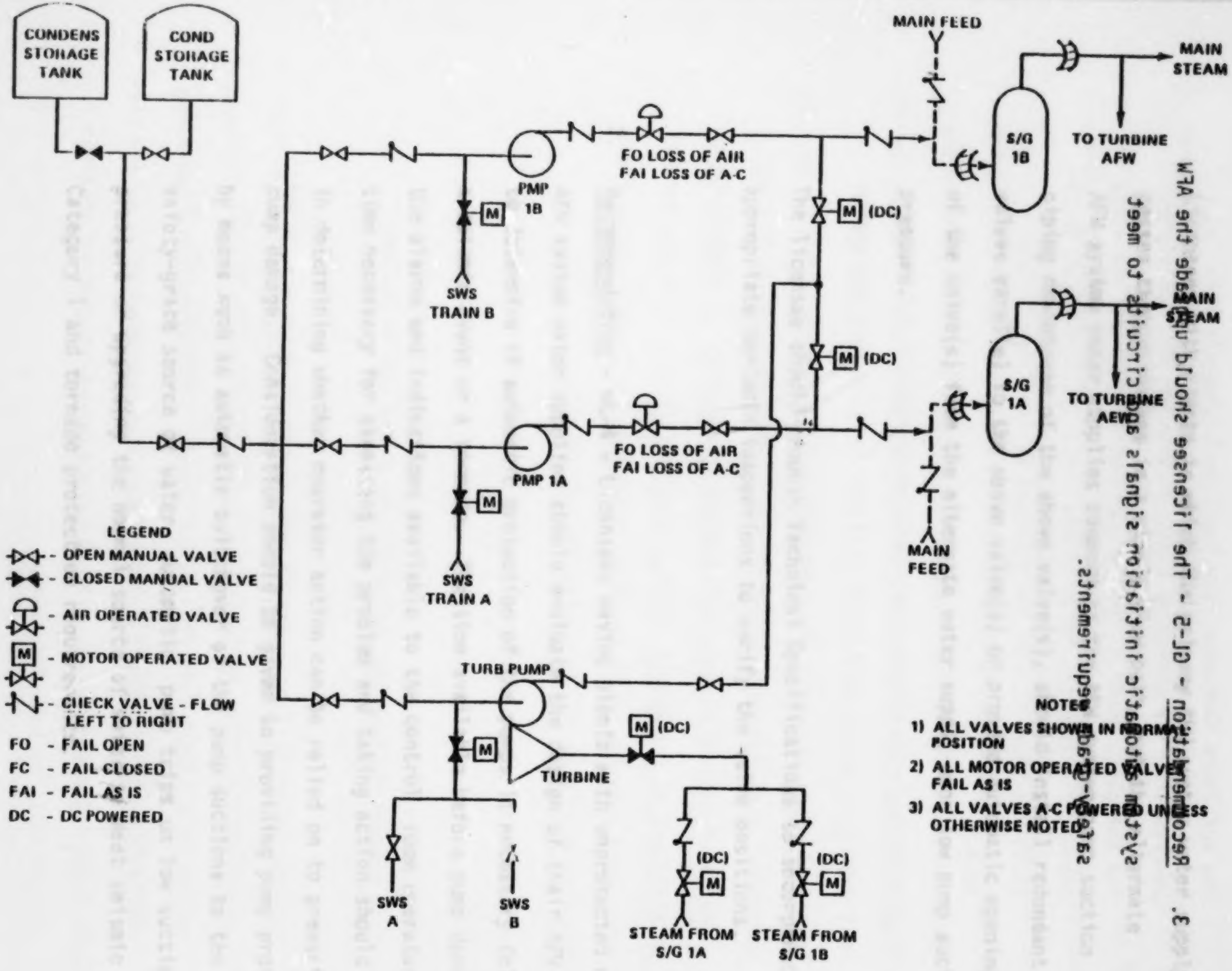
1. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be consider in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

3. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.


CONDENS STORAGE TANK
COND STORAGE TANK
MAIN FEED
MAIN STEAM
S/G 1B
TO TURBINE AFW
PMP 1B
FO LOSS OF AIR
FAI LOSS OF A-C
M (DC)
M
SWS TRAIN B
MAIN STEAM
S/G 1A
TO TURBINE AFW
M (DC)
PMP 1A
FO LOSS OF AIR
FAI LOSS OF A-C
MAIN FEED
M
SWS TRAIN A
TURB PUMP
M (DC)
M
TURBINE
SWS A
SWS B
(DC) M
(DC) M
STEAM FROM S/G 1A
STEAM FROM S/G 1B
LEGEND
- OPEN MANUAL VALVE
- CLOSED MANUAL VALVE
- AIR OPERATED VALVE
M - MOTOR OPERATED VALVE
- CHECK VALVE - FLOW LEFT TO RIGHT
FO - FAIL OPEN
FC - FAIL CLOSED
FAI - FAIL AS IS
DC - DC POWERED
NOTES
1) ALL VALVES SHOWN IN NORMAL POSITION
2) ALL MOTOR OPERATED VALVES FAIL AS IS
3) ALL VALVES A-C POWERED UNLESS OTHERWISE NOTED

Auxiliary Feedwater System
Kewaunee
Figure 1

12. NORTH ANNA 1 AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

X.9 (W) NORTH ANNA UNIT 1

AUXILIARY FEEDWATER SYSTEM

X.9.1 System Description

X.9.1.1 Configuration and Overall Design

The auxiliary feedwater system (AFWS) is designed to supply water to the steam generators (SG) for reactor coolant system sensible and decay heat removal when the normal feedwater system is not available. The AFWS can be utilized during certain periods of normal startup and shutdowns, in the event of malfunctions such as loss of main feedwater flow, loss of offsite power and also in the event of an accident. The AFW system is automatically initiated upon receipt of the following signals: low steam generator level, safety injection, loss of offsite power, and main feed pump trip.

The AFWS is shown in simplified form in Figure 1 attached. The system consists of two motor driven pumps (3A, 3B), and one steam driven turbine pump (2). Each motor driven pump has a design flow of 350 gpm, the turbine driven pump has a design flow of 700 gpm but is orifice limited to 350 gpm when pumping in the normal lineup to SG-A. Taps from each main steam line at a point upstream of the main steam isolation valves provide the source of steam to the turbine. The motor driven pumps are connected to separate emergency power buses

(Class 1E). Normally the pumps take suction from the 110,000 gallon emergency condensate storage tank. This provides 8 hours of operation for decay heat removal. The emergency condensate storage tank is designed to seismic Category I requirements and is protected from tornado missiles. An additional supply of 300,000 gallons is available from a non-seismic condensate make-up storage tank. In addition to the 300,000 gallon supply, an unlimited supply of water is available from the seismic Category I service water and fire protection systems which are supplied from Lake Anna and the spray cooling pond, which is seismic Category I designed.

Referring to Figure 1, each pump is lined up normally to a specific steam generator; pump 3B to SG B, 3A to SG C and 2 to SG A. The pumps can be aligned to other steam generators in the event of line breaks, pump failures, etc., by positioning the manual control valves to suit. AFW flow to steam generator C is normally remote manually controlled by an air operated valve. AFW flow to steam generators A and B is similarly manually controlled by an AC motor operated valve in each supply line. In the event of loss of offsite and onsite AC power, realignment of manual valves is necessary to supply AFW flow to all steam generators from the turbine-driven pump. The instrument air supply system, for the air operated AFW flow control valves, includes a 16.7 cu. ft. accumulator tank charged to 100 psig. This capacity is sufficient to operate the air operated valve(s) from 30 minutes to 8 hours depending on frequency of valve adjustment.

X.9.1.2 Component Design Classification

The turbine pump train and motor pump trains (110,000 gallon tank, pumps, valves, motors, piping, service water and fire protection systems) are seismic Category 1 and tornado missile protected, designed to Quality Group C. (Class 1E for electrical equipment). The 300,000 gallon condensate make up tank is non-seismic.

X.9.1.3 Power Sources

The motor driven pumps and motor operated valves are supplied from the Class 1E A-C emergency buses which may be powered by the diesel generators, 3A from Emergency Bus 1H, 3B from emergency bus 1J. The steam admission valves for the turbine pump are air-operated using DC solenoids and are energized from the emergency battery buses.

Instrumentation and Controls

The instrumentation and Control power supplies are from the 120 VAC vital bus system. There are 4 vital buses, each supplied by an inverter from the 125 VDC power system. The motor driven pump breaker controls are powered from the 125 VDC power system provided from the Class 1E emergency DC buses.

X.9.1.4 Controls

Steam generator level is controlled remote manually from either the Main Control Room (MCR) or the Auxiliary Shutdown Panel (ASP) with safety grade instrumentation provided (level and flow indications).

The valves in the water flow lines to the steam generators consist of three motor operated valves from one header and three air operated valves from the other header. Any pump can supply either header by operating manual valves in the pumps discharge. (For normal alignment refer to Section X.9.1.1.) The air operated valves are normally open and fail open on loss of power, the MOVs are normally open and fail as-is on loss of power. The MOVs or air operated valves are positioned by the operator to maintain proper level in the steam generators.

The steam admission valves to the turbine are air operated, normally closed and fail open. These valves can be controlled from the MCR or ASP.

X.9.1.5 Information Available to the Operator

The following indications are available at both operating stations except as noted.

1. Position indication of the MOVs and air operated valves
2. Flow gpm to each steam generator (Main Control Board (MCB)) only
3. Pump Current and Voltage (MCB only)
4. Steam Pressure to Aux feed pump turbine (MCB Only)
5. Steam Generator Levels
6. Pump Discharge and suction pressures (MCB only)
7. Breaker (motor driven pumps) position
8. Condensate storage tank level emergency

X.9.1.6 Initiating Signals for Automatic Operations

The following signals start the pump motors and open the steam control valves to the turbine:

1. Steam generator water level, low-low in any steam generator (2 out of 3 signals)
2. Safety Injection signal* (Delay of 35-60 seconds on Motor driven pumps)
3. Loss of offsite power
4. Main Feed pump trip (loss of all Main feed pumps)
5. Manual

X.9.1.7 Testing

The systems are tested periodically in accordance with tech spec requirements. The frequency of periodic testing is 31 days. In addition, the particular system is tested in accordance with the technical specifications after performing system maintenance. The systems are tested using the recirculating lines, with various plant parameters noted. (Suction and discharge pressure, etc.) The instrumentation systems are checked periodically, in accordance with the technical specifications, on a per shift, monthly or refueling time frame basis.

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*There is no delay of the S.I. signal to the turbine driven pump control. The reason for the delay in the motor driven pump control circuit is to limit the loads during emergency diesel generator loading.

X.9.1.8 Technical Specifications

A review of the technical specifications indicated that these specifications cover limiting conditions of operation (LCO) and periodic surveillance testing consistent with standard Technical specifications.

X.9.2 Reliability Evaluation

X.9.2.1 Dominant Failure Modes

Successful delivery of feedwater is considered to be the flow of at least 350 gpm to one (or more) of the three steam generators for the transients considered here.

Failure modes of the AFWS were assessed for three types of initiating transients. The dominant failure modes for each transient type are discussed below.

Loss of MFW with Offsite Power Available

The reliability analysis of the North Anna AFWS based on this initiating transient did not identify any single failures or double failures which would fail the entire AFWS. This assessment indicates the dominant AFWS failure mode to be a combination of the failure of both actuation trains to actuate their respective components, coupl with failure of the operator to detect the non-actuation of the system and to manually actuate it.

Loss of MFW with Only Onsite AC Power Available

This transient is somewhat different in character than in the case above, in that reliance for AC power is now on the station diesel generators rather than offsite power. In essence, this adds failure modes such as diesel-generator failure to start to the overall list of failure modes of trains of the AFWS.

The dominant failure modes for the AFWS discussed for the above case are not dependent on the actual source of AC power (i.e., offsite vs. onsite). Thus the probability of the failure mode discussed above should not change. Further, the addition of the diesel-generator failure mode to other train failure modes is not sufficiently important to make the probability of such a combination of modes significant. For these reasons,the AFWS failure probability for this case is still dominated by the coincident loss of both actuation trains, coupled with the failure of the operator to subsequently manually actuate the AFWS.

Loss of MFW with Only DC Power Available

In this case, no AC power (offsite or onsite) is available; the AFWS is thus reduced to the one steam-driven train for feedwater delivery. A number of single failures within this train can fail the AFWS (e.g., hardware failure in the pump and valves, control system failures, etc.). The dominant failure mode for this train is that the train is out of service for maintenance when the transient occurs.

X.9.2.2 Principal Dependencies

The potential for location dependencies was noted during this reliability evaluation, in that some portions of the AFWS were located in common rooms. However, because no location dependencies were found which could potentially affect all trains of the AFWS, these dependencies do not appear to be a significant concern.

XIII.9.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AF system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.9.3.1 Short Term

1. Recommendation GS-4 - Emergency procedures for transferring to alter sources of AFW supply should be available to the plant operators. Th procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources sho take place. The following cases should be covered by the procedures

   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

- The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

2. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

3. Recommendation GS-7 - The licensee should verify that the automatic start AFW signals and associated circuitry are safety

grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

9.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrati that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient cond tions (temperature, humidity) do not exceed environmental qualificati limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

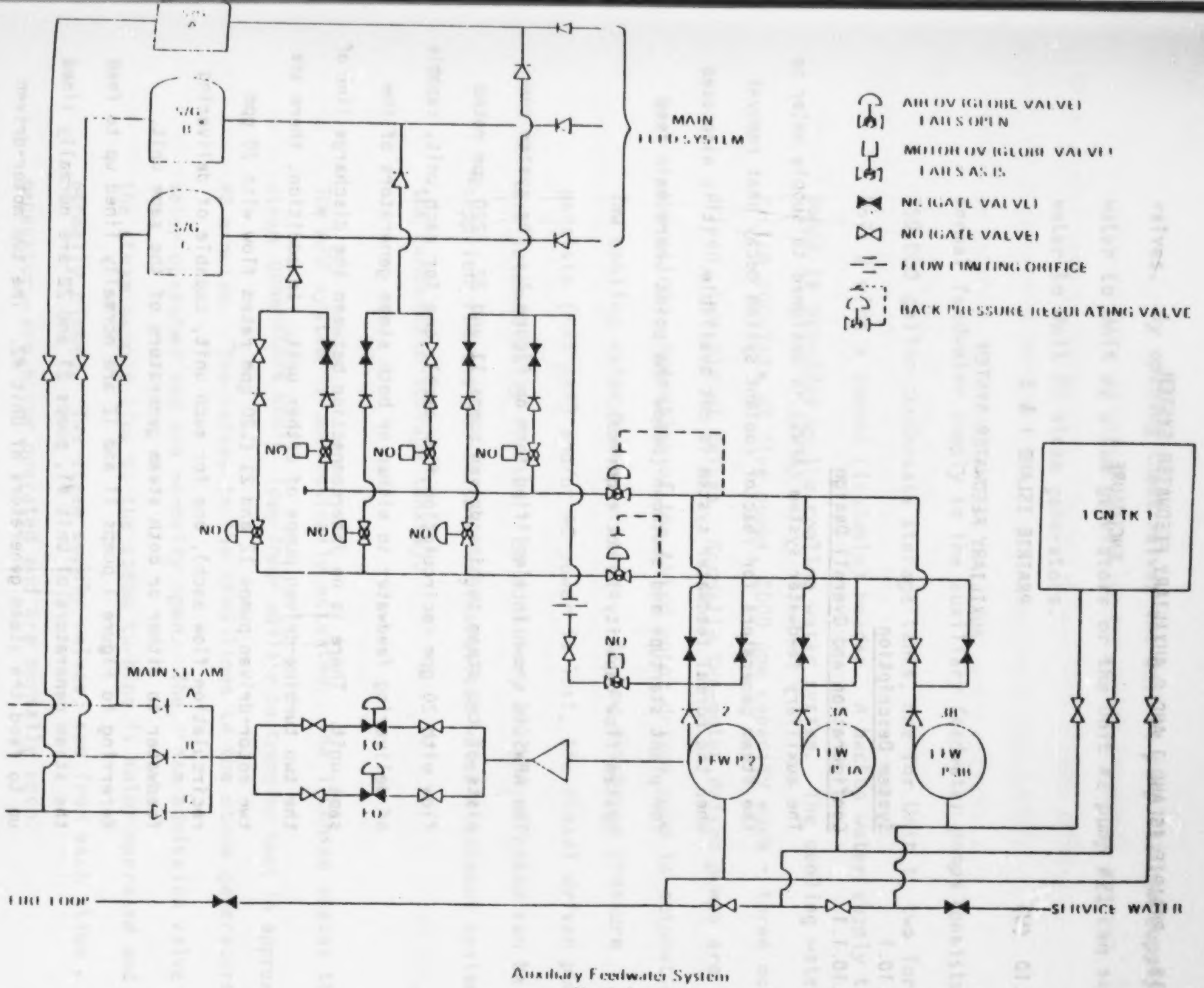
   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the contro room, this operator would re-align the valves in the AFW system trai from the test mode to its operational alignment.

X.9.3.3 Long Term

Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.


S/G A
S/G B
S/G C
MAIN FEED SYSTEM
AIR OV (GLOBE VALVE) FAILS OPEN
MOTOR OV (GLOBE VALVE) FAILS AS IS
NC (GATE VALVE)
NO (GATE VALVE)
FLOW LIMITING ORIFICE
BACK PRESSURE REGULATING VALVE
NO
NO
NO
NO
NO
NO
NO
1 CN TK 1
MAIN STEAM
A
B
C
FO
FO
2
3A
3B
1 FW P2
1 FW P3A
1 FW P3B
FIRE LOOP
SERVICE WATER

Auxiliary Feedwater System
North Anna
Figure 1

13. PRAIRIE ISLAND 1 AND 2 AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

X.10 (W) PRAIRIE ISLAND 1 & 2 AUXILIARY FEEDWATER SYSTEM

X.10.1 System Description

X.10.1.1 Configuration and Overall Design

The auxiliary feedwater system (AFWS) is designed to supply water to the steam generators for reactor coolant system decay heat removal when the normal feedwater system is not available. It is also used for plant startups and shutdown (below the point where main feed system flow capacity is not required).

The AFWS is shown in simplified form on Figure 1. The system consists of two steam turbine-driven pumps 11 and 22, (220 gpm rated flow with 20 gpm recirculation flow each), one for each unit, capable of delivering feedwater to either or both steam generators of the same unit. There is no interconnection between the discharge line of the two turbine-driven pumps of either unit. In addition, there are two motor-driven pumps 12 and 21 (220 gpm rated flow with 20 gpm recirculation flow each), one for each unit, capable of delivering feedwater to either or both steam generators of the same unit. Referring to Figure 1 pumps 11 and 12 are normally lined up to feed the steam generators of Unit #1, pumps 21 and 22 are normally lined up to feed the steam generators of Unit #2. The two motor-driven pump discharge headers are interconnected by two normally closed valves. By opening these valves the Unit #1 pump #12 can supply water to Unit #2 steam generators or the Unit #2 pump #21 can supply water to Unit #1 steam generators.

Normal feedwater supply to the auxiliary feedwater pumps consists of 150,000 gallon condensate storage tanks, one for Unit 1, two for Unit 2 with a common (isolable) header. A backup water supply to the pumps is provided by the cooling water system. The cooling water system consists of five pumps, 2000 gpm capacity each - three motor driven, two diesel driven. Normally two-motor driven pumps are operating. Actuation of the third motor driven pump is automatic on low cooling water header pressure. If low discharge pressure persists (~75 psi) and/or AC power is lost, the diesel driven pumps are automatically started. In addition, 260 gpm of water can be supplied via the non-seismic demineralized water treatment system to the condensate storage tank(s).

The AFW system is automatically actuated. The licensee states that the steam generators would lose their ability to transfer heat in approximately 40 minutes. The valves in the AFWS lines to the steam generators are motor operated and are normally open. The steam admission valve in the steam supply line to the steam turbine is motor operated and is normally closed. Two steam supply valves, one from each steam generator, are motor operated and are normally open.

X.10.1.2 Component Design Classification

The turbine pump trains and motor pump trains (pumps, valves, motors, piping) are seismic Category I, tornado missile protected and designed to Quality Group I. Electrical equipment is designed as Class IE.

The sources of water and associated piping are classified as follows:

1. Condensate storage tank (Unit 1) } Type 3 (Non-seismic)
   Condensate storage tanks (Unit 2)
2. Cooling Water System - Seismic Category I, Tornado Missile Protected
3. Suction Piping
   from condensate storage tanks - Class 2B (non-seismic)
   from cooling water system (seismic - Category 1)
4. Demineralized water treatment system (non-seismic)

X.10.1.3 Power Sources

The motor driven pumps are supplied from the Class IE emergency buses, (Bus #16 - Train B for #12 pump, Bus 26 - Train A for #21 pump) Motor operated valve (MOV) power is from the emergency buses on a train basis. The emergency buses are capable of being powered from the diesel generators. Steam for the turbine driven pumps is supplied from each steam generator of the respective reactor unit.

The instrumentation and control power supplies are from the 120 VAC vital bus system. There are four vital buses/unit, each supplied by an inverter connected to the 480 VAC emergency bus and the 125 VDC power system. The motor driven pump breaker controls are powered from the 125 VDC control batteries which are charged by battery chargers connected to the 480 VAC emergency buses.

X.10.1.4 Instrumentation and Control

Controls

Any of the MOV's can be controlled from either the Main Control Room or the Hot Shutdown Panel (local station).

Steam generator level is controlled by positioning the MOV's in the flow discharge lines. Level and flow indication is provided for operator information.

The valves are motor operated and fail as-is on loss of power.

X.10.1.4.1 AFW System Information Available to the Operator (At both remote and local stations except as noted)

1. MOV Position
2. S/G Level and pressure indication (alarm-control room only)
3. S.I. Ready Panel-abnormal valve position and AFW pump operability status-(control room only)
4. Discharge Pressure
5. Discharge Flow
6. CST Level Indication (low level alarm-control room only)

X.10.1.4.2 Initiating Signals for Automatic Operation

The following signals start the pump motors and open the steam admission control valve to the turbine of the affected unit:

1. Low-low water level in either steam generator
2. Trip of both main feedwater pumps
3. Safety injection
4. Undervoltage of both 4.16 kv normal buses (turbine driven pump only)
5. Manual

X.10.1.6 Testing

Auxiliary feed system surveillance tests are required on a monthly and refueling interval in accordance with tech spec requirements. The monthly tests involve (a) stroking MOV's and observing stem travel and (b) pump curve point check. The test is performed by shutting the appropriate pump discharge valves and recirculating back to the CST. After the test, the valves are positioned to normal lineup and all valve positions are verified.

X.10.1.7 Technical Specifications

The present limiting condition of operation (LCO) permits one unit operation with one motor-driven pump operable and either one turbine or one motor driven pump operable and if failure occurs and is not fixed in 48 hours - go to cold shutdown. Two unit operation is permitted with all four AFW pumps operable. If a failure occurs and repair is not completed within 7 days so that the four pump requirement is met, one unit must be taken to cold shutdown.

X.10.2 Reliability Evaluation Results

X.10.2.1 Dominant Failure Modes

Failure modes of the AFWS were assessed for three types of initiating transients. The dominant failure modes for each transient type are discussed below.

It should be noted that the failure modes discussed below as dominant presume LCOs of 48 hours on all AFWS trains. Currently, the LCO allows both the turbine-driven trains to be unavailable indefinitely when only one of the two units is operating. See Recommendation GS-1.

Loss of MFW with Offsite Power Available

A dominant failure mode of the AFWS for this transient is assessed to be the blockage of flow to the two steam generators due to inadvertent closure of two manual valves in the pump discharge lines inside containment. These valves could possibly be closed prior to the AFWS demand because of, for example, a personnel error in failing to reopen them after maintenance on the AFWS. Because these valves do not have remote position indication and are located inside the containment, there could be a considerable delay in gaining access to and reopening of the valves after an AFWS demand. However, the licensee states that any inadvertent closure of these valves would be detected prior to reactor startup or at least before the reactor exceeds 2% power since (a) plant startup procedures require a valve alignment check to verify the AFWS flow path and (b) the AFWS is used during normal plant startup to maintain steam generator water level before initiating operation of the main feedwater system after reaching 2% reactor power; thus inoperability of the AFWS would have been detected before proceeding further. See Recommendations GS-2 and GS-6.

Loss of MFW with Only Onsite AC Power Available

This transient is very similar to the transient discussed above. Additional failure modes related to the onsite AC power system were considered; however, these did not have a significant impact. As such, a dominant failure mode for the case described above (closure of two manual valves in the AFWS discharge lines inside containment) is also considered to be dominant for this transient.

Loss of MFW with Only DC Power Available

In this transient no AC power (onsite or offsite) is available, so that the AFWS is reduced to one steam-driven pump train. The dominant failure mode of this train in such a transient is assessed to be failure of the operator to open the normally closed steam-admission valve before the steam generator water level decreases to the point where it loses its ability to transfer heat. This valve is motor-operated and is normally powered from either offsite AC power or from the diesel-generators. Since neither of these power sources is available in this transient, local, manual opening of the valve would be required.

X.10.2.2 Principal Dependencies

The principal dependency found in this evaluation is the common-cause failure of all trains due to closure of the manual valves in the two AFWS discharge lines.

The second significant dependency found is the dependence on AC power to run the turbine-driven pump train of the AFWS.

Because of physical separation of the AFWS pumps, location dependencies do not appear to be significant in this plant.

The AFWS pumps require cooling from the plant cooling water system. However, since this system can be run from offsite or onsite AC power supplies, and also has separate diesel-driven pumps, this potential common cause failure does not appear to be of significance.

.10.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant specific) identified in this section represent actions to improve AFW systems reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL, and plant specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.10.3.1 Short Term

1. Recommendation GS-1 - The Technical Specification LCO for one unit operation allows the turbine-driven pump train of that unit to be unavailable indefinitely. Consequently, the plant could not provide AFW flow in the event of loss of offsite and onsite AC power. The licensee should propose modifications to the Technical Specifications to limit the time that a turbine-driven pump train can be inoperable during single unit operation. The licensee should update the Technical Specification LCO for both one and two unit operation to conform with current standard Technical Specifications; namely 72 hours and 12 hours for the outage time limit and action time.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow, including the manual valves V12 and V25 inside containment. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

3. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

4. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearing may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the

emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

5. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendation GS-7 - The licensee should verify that the automatic start AFW signals and associated circuitry are safety grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet

the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.10.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletin and Orders Tas Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and a low level alarm in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   ". Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

.10.3.3 Long Term

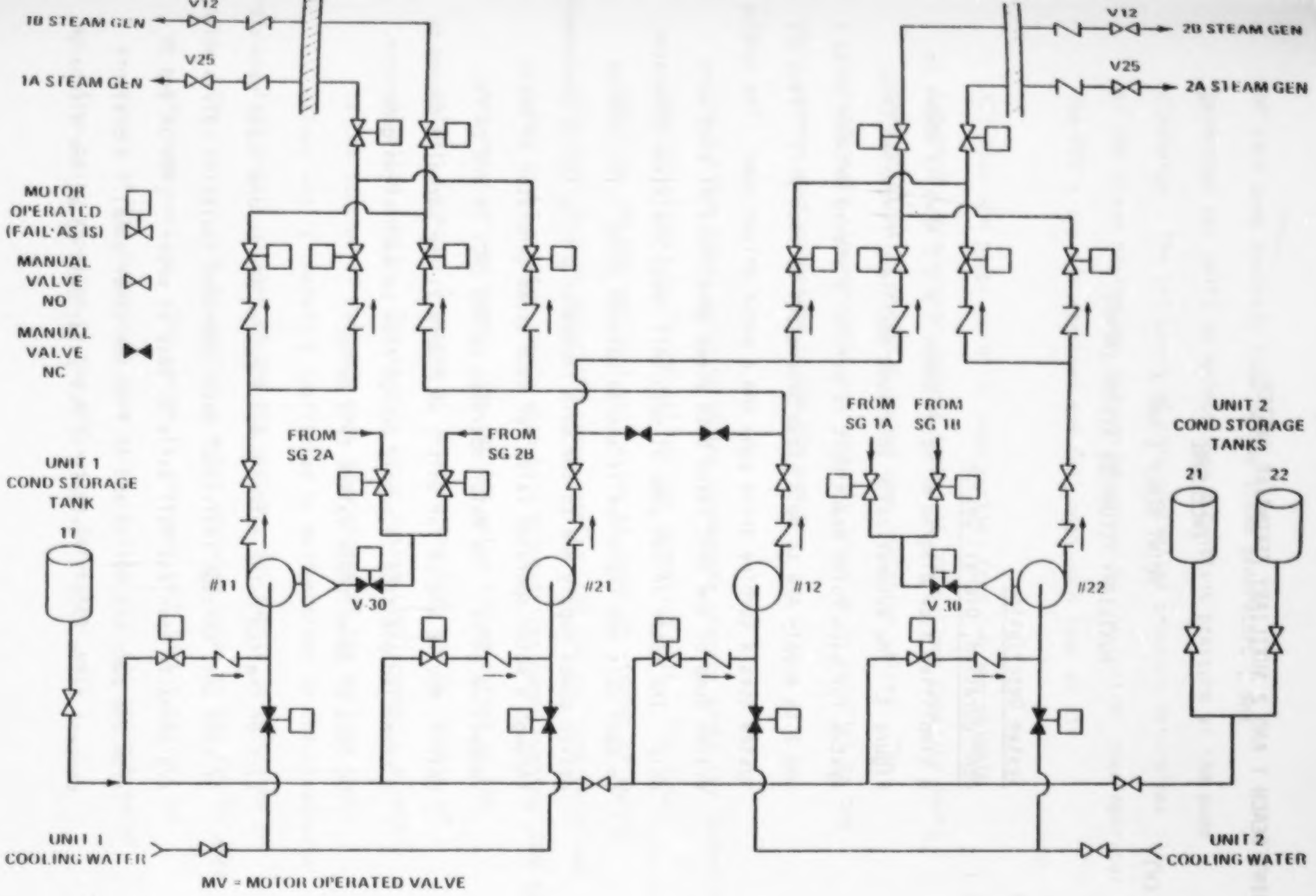
Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being

operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

2. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pump is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available for the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trip on low suction pressure or upgrading the normal source of water to meet seismic Category I tornado protection requirements.

3. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.


1B STEAM GEN
V12
1A STEAM GEN
V25
V12
2B STEAM GEN
V25
2A STEAM GEN
MOTOR OPERATED (FAIL AS IS)
MANUAL VALVE NO
MANUAL VALVE NC
FROM SG 2A
FROM SG 2B
FROM SG 1A
FROM SG 1B
UNIT 1 COND STORAGE TANK
11
UNIT 2 COND STORAGE TANKS
21
22
#11
V-30
#21
#12
V 30
#22
UNIT 1 COOLING WATER
UNIT 2 COOLING WATER
MV = MOTOR OPERATED VALVE

Auxiliary Feedwater System
Prairie Island
Figure 1

14. POINT BEACH 1 AND 2 AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

X.11 (W) POINT BEACH 1 AND 2

AUXILIARY FEEDWATER SYSTEM (AFWS)

X.11.1 System Description

X.11.1.1 Configuration, Overall Design

A simplified flow diagram of Point Beach 1 and 2 AFWS is shown in Figure 1. The automatically initiated auxiliary feedwater (AFW) system for each Point Beach Unit is partially shared between units 1 and 2 to supply AFW to both steam generators of each unit. Each AFW system uses a turbine drive pump and a motor driven pump. The turbine driven pump of one unit feeds both steam generators of that unit only. The motor driven pump of each unit feeds one steam generator in each unit and therefore is shared between units. The turbine driven pumps supply AFW to the main feedwater piping inside containment through a motor operated valve for each steam generator of their respective units. The motor operated valves (MOV 1A and 2A for Unit 1, MOV 1B and 2B for Unit 2 on Figure 1) are normally opened to a throttled position to supply design flow to each steam generator. On loss of power these valves fail as-is.

Each of the two motor driven pumps supplies AFW to one steam generator of each unit through individual motor operated isolation valves which are normally open and fail as-is on loss of power. (MOV 3A and 3B from one pump and MOV 4A and 4B from the other pump). A pressure control valve (PCV-1 for Unit 1, PCV-2 for Unit 2) at the discharge of each pump controls flow to two steam generators (one generator per unit) by maintaining a constant pressure at the pump discharge. The set point of this controlled pressure determines flow to the steam generators and can be varied by the control room operator. The PCV's are air operated and fail open upon loss of air.

All four AFW pumps normally take suction from two non-seismic Category I condensate storage tanks (45,000 gallons capacity each) through manually operated locked open isolation valves. The condensate storage tanks are normally lined up in parallel to the common suction header of the AFW pumps.

The minimum total capacity of the condensate storage tanks (by Technical Specifications) is 10,000 gallons per operating unit. The total capacity (20,000 gallons) will allow at least 25 minutes of supply with both turbine drive AFW pumps running (400 gpm per turbine-driven pump) or 50 minutes supply with both motor-driven pumps running (200 gpm per motor driven pump). The service water system serves as the seismic Category I source of water to the AFWS and is capable of unlimited supply. The service water system (SWS) connects directly to the suction of each AFW pump down-stream of the suction check valves and is therefore unaffected by malfunctions in the condensate tank supply portion of the AFW system. SWS supply is initiated in the control room by opening a motor operated valve in the SWS to each AFW pump suction. The system is arranged such that a failure of either of the two diesel generators on site will not prevent water from being supplied to the AFW system for either unit.

Since all valves in the flow path to the steam generators are normally open and fail as-is (with exception of PCV-1 and 2 which fail open) a loss of A-C or D-C power does not require valve manipulation. The motor operated steam valves at the inlet to the turbines (MS-1A and 2A for Unit 1, MS-1B and 2B for Unit 2) are D-C motor operated valves and will automatically open in the event of a loss of all A-C power.

In the event of an unisolable main steam or feedwater line break coincident with a worst case single active failure, operator action within the control room will isolate AFW flow to the affected steam generator and assure flow to the unaffected steam generator. The licensee estimates >30 minutes to boil dry.

A break anywhere in the auxiliary feedwater system discharge piping would not prevent automatic AFW flow to at least one steam generator on demand. A single active failure coincident with a break could disable automatic AFW to both steam generators, depending on break location. In either case, breaks could be isolated by operator action within the control room.

1.1.2 Component Design Classification

All pumps, valves, piping, instrumentation and controls associated with the auxiliary feedwater system (except Condensate Storage Tanks) are designed safety Class I which includes seismic Category I requirements.

The condensate storage tank and associated instrumentation are not designed to safety grade requirements. The piping from the tank to the auxiliary feedwater system is classified as safety class I which includes seismic Category 1 requirements.

X.11.1.3 Power Sources

Power sources for all instrumentation and controls are taken from the emergency buses which are supplied by the safety related diesel generators or safety related station batteries. Steam generator water level control and the automatic initiation system are designed as a safety related system, including seismic Category I.

Each motor driven pump and associated instrumentation and controls are powered by a separate diesel-generator, such that a failure of one diesel generator will only disable one motor driven train.

The turbine driven pump for each unit receives steam from both steam generators of its respective unit through parallel d-c motor operated isolation valves. The parallel valves are powered from separate D-C buses such that a loss of one d-c system will not prevent operation of either turbine driven pump.

X.11.1.4 Instrumentation and Controls

X.11.1.4.1 Controls

All controls for the active components of the auxiliary feedwater system can be operated from the control room. Normally steam generator level is controlled in the control room by adjusting the pressure set point of the pressure control valves at the discharge of each motor driven pump. If it is necessary to control turbine pump flow for level control, the motor operated valves in the discharge lines from the turbine driven pump each steam generator can be throttled from the control room.

Each control actuator in the control room is located in a basic system layout (MIMIC Bus) to help identify the control switch function in addition to the identifying name plate.

X.11.1.4.2 Information Available to the Operator

I. Alarms

- a) Hi/Lo Steam Generator Level
- b) Low Level - Condensate Storage Tank
- c) Service Water System Header Pressure Low

II. Indication

- a) Steam Generator Level
- b) Condensate Storage Tank Level
- c) AFW pump discharge pressure
- d) Service Water Header pressure
- e) Valve Position Indication - All Active Valves
- f) Pump Running Lights - Motor Drive
- g) Pump Breaker Trouble Light - (Did Not Close on Demand)
- h) Pressure Set Point - Pressure Control Valve

All valve position indicators are located with their respective controls on the "MIMIC Board" such that the valves are readily identified.

X.11.1.4.3 AFW Initiating Signals

I. Turbine Pumps

- a) Lo-Lo Level in both S/G's of its respective unit - automatic
- b) Loss of both 4 KV busses (Supply reactor coolant Pumps) - automatic
- c) Manual - From Control Room

II. Motor Driven Pumps

- a) Lo-Lo Level in any one S/G of either unit - automatic
- b) Trip of both Main Feed Pump - either unit - automatic
- c) Safety Injection Signal - either unit - automatic
- d) Manual - from Control Room

X.11.1.5 Testing

1) Valve position is verified monthly
2) Service Water System supply valves are cycled monthly
3) Operational tests of AFW pumps are performed monthly by verifying pump suction and discharge pressure (Tests are staggered)
4) Flow verification tests from condensate tanks to S/G's are performed at each refueling or whenever in cold shutdown (Not more frequently than quarterly)

5) Automatic initiation of the AFW system is verified during each refueling.

6) Control and initiating circuits are tested with each pump and valve test

7) Following maintenance on the system, an operational test is performed to bring the system back in service.

X.11.1.6 Technical specifications

A. When the reactor coolant is heated above 350°F the reactor shall not be taken critical unless the following conditions are met:

1a. Two Unit Operation - Three of the four auxiliary feedwater pumps are operable.

1b. Single Unit Operation - Either the turbine driven pump associated with that unit together with one of the two motor driven pumps or both motor driven pumps must be operable.

2. A minimum of 10,000 gallons of water per operating unit in the condensate storage tanks and an unlimited water supply from the lake via either leg of the plant service water system.

3. System piping and valves required to function during accident conditions directly associated with the above components must be operable.

B. During power operation, the requirements are modified to allow the following components to be inoperable for a specified time.

If the system is not restored to meet the above requirements within the time period specified the appropriate reactor(s) shall be placed in the hot shutdown condition. If they are not satisfied within an additional 48 hours, the appropriate reactor(s) shall be cooled down to less than 350°F.

1. Two Unit Operation - One of the three operable auxiliary feedwater pumps may be out-of-service provided a pump is restored to operable status within 24 hours.
2. Single Unit Operation - One of the two operable auxiliary feedwater pumps may be out-of-service provided a pump is restored to operable status within 24 hours.

X.11.2 Reliability Evaluation

X.11.2.1 Dominant Failure Modes

The dominant failure modes are expressed for three transient situations and two operational configurations, single unit operation and double unit operation.

Limiting conditions for single unit operation are a single motor-driven pump and associated turbine driven pump operable or both motor driven pumps operable. Any one can be out of service for 24 hours.

Limiting conditions for double unit operation are three of four auxiliary feedwater pumps operable. Any one can be out of service for 24 hours.

LOFW with Only DC Power Available

Single Unit Operation

The dominant failure contributor is loss of both motor-driven pumps and subsequent failure of the turbine driven pump due to loss of service water (AC) cooling to steam turbine pump bearing oil.

Double Unit Operation

Same failure as single unit operation.

X.11.2.2 Interdependencies

The principal noted dependency is the design for AC cooling of the turbine driven pumps.

X.11.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.11.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumention can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

4. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

- The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

- The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

5. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:
   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system shou be modified in the short-term to meet the functional requirements liste

11.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator actions, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions

(temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the contro room, this operator would re-align the valves in the AFW system trai from the test mode to its operational alignment.

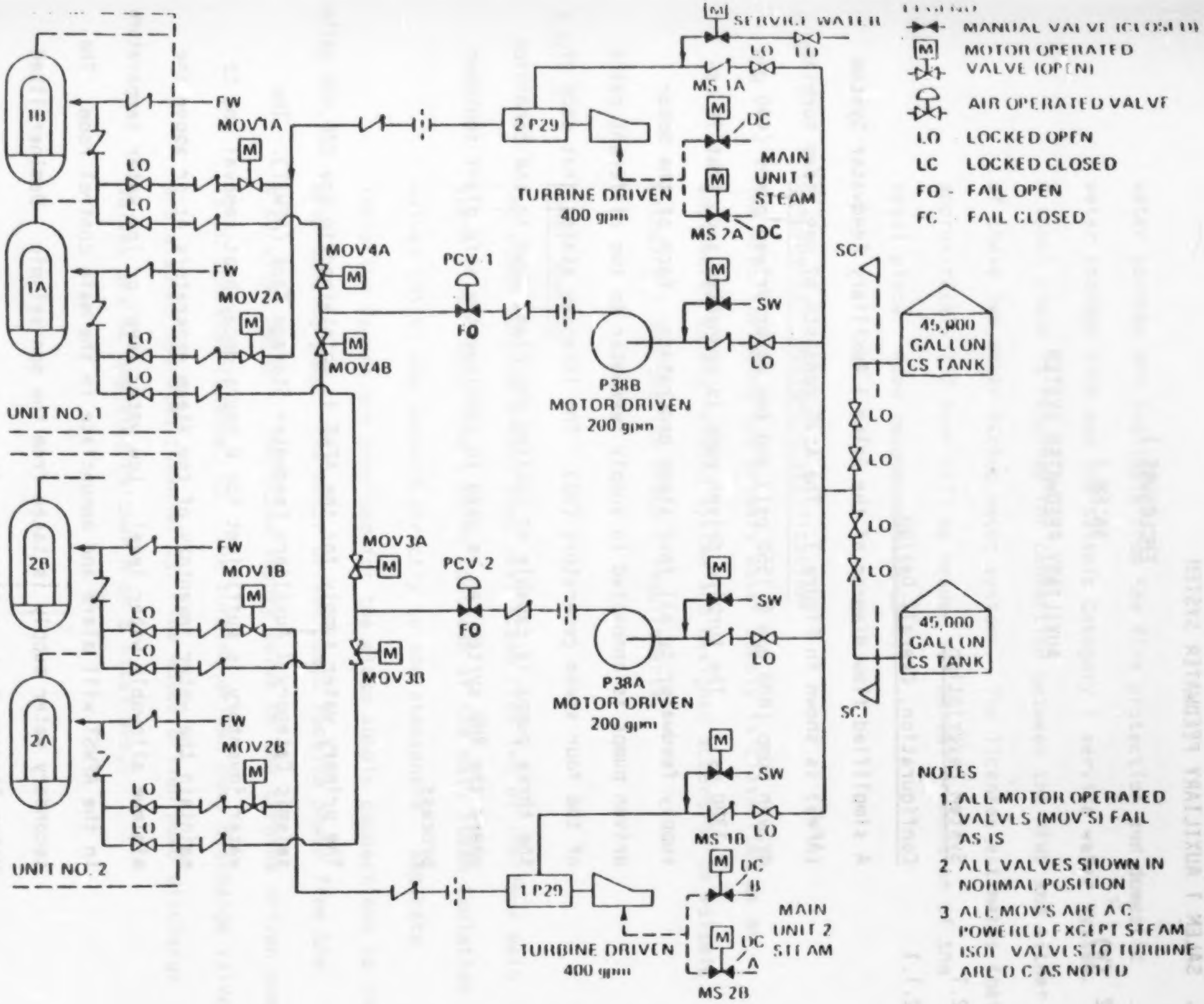
X.11.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating curre is acceptable.

2. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

3. Recommendation-GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.



SERVICE WATER
LEGEND
MANUAL VALVE (CLOSED)
M MOTOR OPERATED VALVE (OPEN)
AIR OPERATED VALVE
LO LOCKED OPEN
LC LOCKED CLOSED
FO - FAIL OPEN
FC - FAIL CLOSED
1B
1A
2B
2A
FW
MOV1A
MOV2A
MOV4A
MOV4B
MOV3A
MOV3B
MOV1B
MOV2B
LO
UNIT NO. 1
UNIT NO. 2
2 P29
1 P29
TURBINE DRIVEN 400 gpm
MS 1A
MS 2A
DC
MAIN UNIT 1 STEAM
MAIN UNIT 2 STEAM
MS 1B
MS 2B
PCV 1
PCV 2
FO
P38B MOTOR DRIVEN 200 gpm
P38A MOTOR DRIVEN 200 gpm
SW
SCI
45,000 GALLON CS TANK
NOTES
1. ALL MOTOR OPERATED VALVES (MOV'S) FAIL AS IS
2. ALL VALVES SHOWN IN NORMAL POSITION
3. ALL MOV'S ARE A C POWERED EXCEPT STEAM ISOL VALVES TO TURBINE ARE D C AS NOTED

Auxiliary Feedwater System
Pt Beach 1 & 2
Figure 1

15. SALEM 1 AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

X.12 (W) SALEM 1

AUXILIARY FEEDWATER SYSTEM

X.12.1 System Description

X.12.1.1 Configuration, Overall Design

A simplified flow diagram of the Salem 1 Auxiliary Feedwater System (AFWS) is shown in Figure 1. The AFWS consists of one steam turbine driven pump (880 gpm @ 1550 psi) and two motor driven pumps (440 gpm @ 1300 psi). The turbine driven pump is connected such that it can supply feedwater to all four steam generators. Each of the motor driven pumps is connected to supply feedwater to two different pairs of the four steam generators (SG). The licensee states that each of the three pumps is capable of cooling the plant down to the condition where the RHR system can be used to continue the safe plant shutdown process.

The primary water supply for the AFWS is maintained in the 220,000 gallon seismic Category I Auxiliary Feedwater Storage Tank (AFWST). The water inventory is sufficient for 8 hours decay heat removal and to maintain the water inventory of the steam generators at or above the minimum allowable water level. Low water level or low water temperature in the AFWST will alarm and annunciate in the main control room. The secondary water supply is taken from the non-seismic Demineralized Water Storage Tanks (500,000 gallons each). In addition, two other backup water sources are available from the fire protection and domestic water storage tank and the seismic Category I service water system. A spool piece is required for connection between the AFWS and either of these two other backup water systems. The licensee estimates that approximately 1/2 hour will be needed for manual connection of the spool piece. (see recommendation GS-4)

.12.1.2 Components - Design, Classification

The AFWS including the primary water supply is classified as an engineered safety-related system and designed according to seismic Category 1 and Quality Group C requirements.

.12.1.3 Power Sources

The steam turbine driven pump uses steam from two of the four main steam lines taken from a point upstream of the main steam isolation valves (MSIV) and exhausts directly to the atmosphere. Separate isolation valves are provided for the steam supply connections to the AFWS turbine pump. The motor driven pumps receive power from the 4 KV vital buses. The steam supply valve to the turbine driven pump is DC operated (Channel C). The turbine driven pump discharge valves are also DC operated (Channel C). The motor driven pump discharge valves are AC operated (either A or B vital bus).

X.12.1.4 Instrumentation and Controls

X.12.1.4.1 Controls

The instrumentation and controls within the AFWS have been designed as seismic Category I and IEEE 279 components. The SG level is remote manually controlled in the main control room. When the level in any SG is ≤ 10% on the narrow range instrumentation, the feedwater flow to the SG is limited to ≤ 1.2 in/min. All power operated valves can be manually controlled from the control room; however, on loss of AC or DC power, these valves can be operated locally.

X.12.1.4.2 Information Available to Operator

The information available to the operator includes pump operability (suction pressure, discharge pressure and discharge flow), AFWST discharge pressure, AFWST level and temperature, steam generator steam flow, steam generator water level and control valve position indication. The system instrument designation on Figure 1 and the associated function are listed below.

Instruments

PD-1043	Auxiliary Feed Storage Tank Discharge Pressure.
PA-1676	Auxiliary Feed Storage Tank Discharge Pressure.
PL-1675	Auxiliary Feed Storage Tank Discharge Pressure.
PA-1039	No. 11 Auxiliary Feedpump suction pressure.
PL-3448	No. 11 Auxiliary Feedpump suction pressure.
PT-1677	No. 11 Auxiliary Feedpump suction pressure.

PA-1040	No. 12 Auxiliary Feedpump suction pressure.
PL-3447	No. 12 Auxiliary Feedpump suction pressure.
PT-1683	No. 12 Auxiliary Feedpump suction pressure.
PA-1041	No. 13 Auxiliary Feedpump suction pressure.
PL-3446	No. 13 Auxiliary Feedpump suction pressure.
PT-1685	No. 13 Auxiliary Feedpump suction pressure.
PA-3450	No. 11 Auxiliary Feedpump discharge pressure.
PA-1081	No. 11 Auxiliary Feedpump discharge pressure.
PL-1678	No. 11 Auxiliary Feedpump discharge pressure.
PA-3449	No. 12 Auxiliary Feedpump discharge pressure.
PA-1082	No. 12 Auxiliary Feedpump discharge pressure.
PL-1684	No. 12 Auxiliary Feedpump discharge pressure.
PA-1083	No. 13 Auxiliary Feedpump discharge pressure.
PL-1686	No. 13 Auxiliary Feedpump discharge pressure.
FA-1037	No. 11 Auxiliary Feedpump discharge flow.
FA-1038	No. 12 Auxiliary Feedpump discharge flow.
FA-1087	No. 11 Steam Generator steam flow.
FA-1091	No. 12 Steam Generator steam flow.
FA-1095	No. 13 Steam Generator steam flow.
FA-1097	No. 14 Steam Generator steam flow.
TD-3608	Auxiliary Feedwater Storage Tank temperature.
LD-2955	Auxiliary Feedwater Storage Tank level.
LD-3601	Auxiliary Feedwater Storage Tank level.
LL-3443	Auxiliary Feedwater Storage Tank level.
LA-1688	Auxiliary Feedwater Storage Tank level.

X.12.1.4.3 Initiating Signals for Automatic Operation

The motor driven pumps will start automatically on any of the following conditions: loss of offsite power, loss of main feedwater flow, safeguards sequence signal, or low-low level signal from any one of the four steam generators. When either of these pumps is started, a start indication is shown on the status panel in the control room, as well as the remote control station and local indication at its local AFW control panel. The turbine driven pump is started by any one of the following conditions: loss of offsite power, low-low level in two of the four steam generators or undervoltage signal in the RCP group buses.

X.12.1.5 Testing and Technical Specifications

The licensee indicated that after each AFW train maintenance outage, the train would be flow tested to the SG to ensure system flow path alignment. Technical Specifications, Auxiliary Feedwater Systems, Surveillance Requirements 4.7.1.2 provide for certain testing at least once per 31 days in accordance with Station Procedure SP(O) 4.7.1.2(a). Testing the 13 steam driven feed pump does not prevent the two motor driven feed pumps or their associated flow paths from performing their intended function. Technical Specifications Section 4.0.5, in-service testing of pumps, requires periodic testing of each motor driven feed pump in accordance with Station Procedure SP(O) 4.0.5-P. Testing of either motor driven (11 or 12) feed pump does not prevent the remaining motor driven feed pump or the steam driven feed pump from performing its individual safety function.

Surveillance Procedure SPO(0) 4.0.5-P Precautions 3.0 states "Do Not Test More Than One Pump At A Time," and Technical Specifications Action Statement 3.7.12 does not permit more than 1 pump to be out of service for more than 72 hours, otherwise be in Hot Shutdown within the next 12 hours.

12.2 Reliability Evaluation Results

12.2.1 Dominant Failure Modes Identified

Normally, any one of three subsystems supplying their pump capacity to at least 2 of the 4 steam generators can provide for adequate decay heat removal (given those three transient events considered).

Presently, however, the flow from the AFWS is throttled back to reduce the potential for occurrence of water hammer due to rapid condensation of steam in the steam generator associated with feedwater addition when the SG water level has dropped below the feed ring. Because of this initially throttled operation of the AFWS (≤ 1.2 in/min), the operator must take steps shortly after AFWS actuation to increase pump flow and the rate of fill of the SGs until the desired water level is reestablished in the SGs.

This throttle back procedure 1) serves to reduce the installed AFWS capacity; 2) it reduces the initial flow capacity that can be claimed to exist for the AFWS designs; 3) it interposes the operator initially into the operation of the AFWS; 4) it creates a risk of delayed refill of the SGs thereby increasing the chance of operation (and sticking open) of the PORVs on the pressurizer; and 5) it affects the overall availability that might otherwise be estimated to exist for the Salem Unit 1 AFWS design. For those reasons, we conclude that the need to maintain such AFWS throttle-back procedures should be reassessed by the licensee.

The following failure modes were found to dominate the demand unavailability of the Salem Unit 1 AFWS.

X.12.2.1.1 Loss of Feedwater (LOFW) with Offsite AC Available

The single manual valve (IAFI) in the suction line from the primary water storage tank was assessed to be the dominant fault contribution ($\lesssim$ 70%) for the Salem 1 AFWS. Presently this manual value is not locked open and if inadvertently closed would result in delay or possibly failure of the AFWS on demand. The pumps could be damaged unless prompt operator action was taken to shut the pumps off and take subsequent actions to reestablish a water supply to the AFWS.

The independent failure of both the steam turbine pump subsystem and either of the motor driven pumps subsystems comprised $\lesssim$ 30% of the next level of dominant fault contributors. These fault contributions would diminish somewhat (such as a factor of two) if the throttle-back procedures were removed.

LOFW with Onsite AC Available

The dominant failure modes for this transient are essentially the same as those described above.

LOFW with Only DC Power Available

For this transient event, the electric AFWS pumps would be unavailable but feedwater to the SG would be automatically provided by the single steam turbine-driven pump. (No AC dependencies were identified for the steam driven portion of the AFWS.) The dominant fault contributions would be those associated with failure of the turbine-driven pumps or the unavailability of this subsystem due to maintenance outage.

12.2.2 Principal Dependencies Identified

The principal dependencies identified were those associated with the single manual valve (IAFI) in the AFWS suction line to the principal water supply tank and the throttle-back practice currently being used. Both of these dependencies give rise to potential human errors that disable the availability of the AFWS.

X.12.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.12.3.1 Short Term

1. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

2. Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

3. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verifv that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.12.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.12.3.3 Long-Term

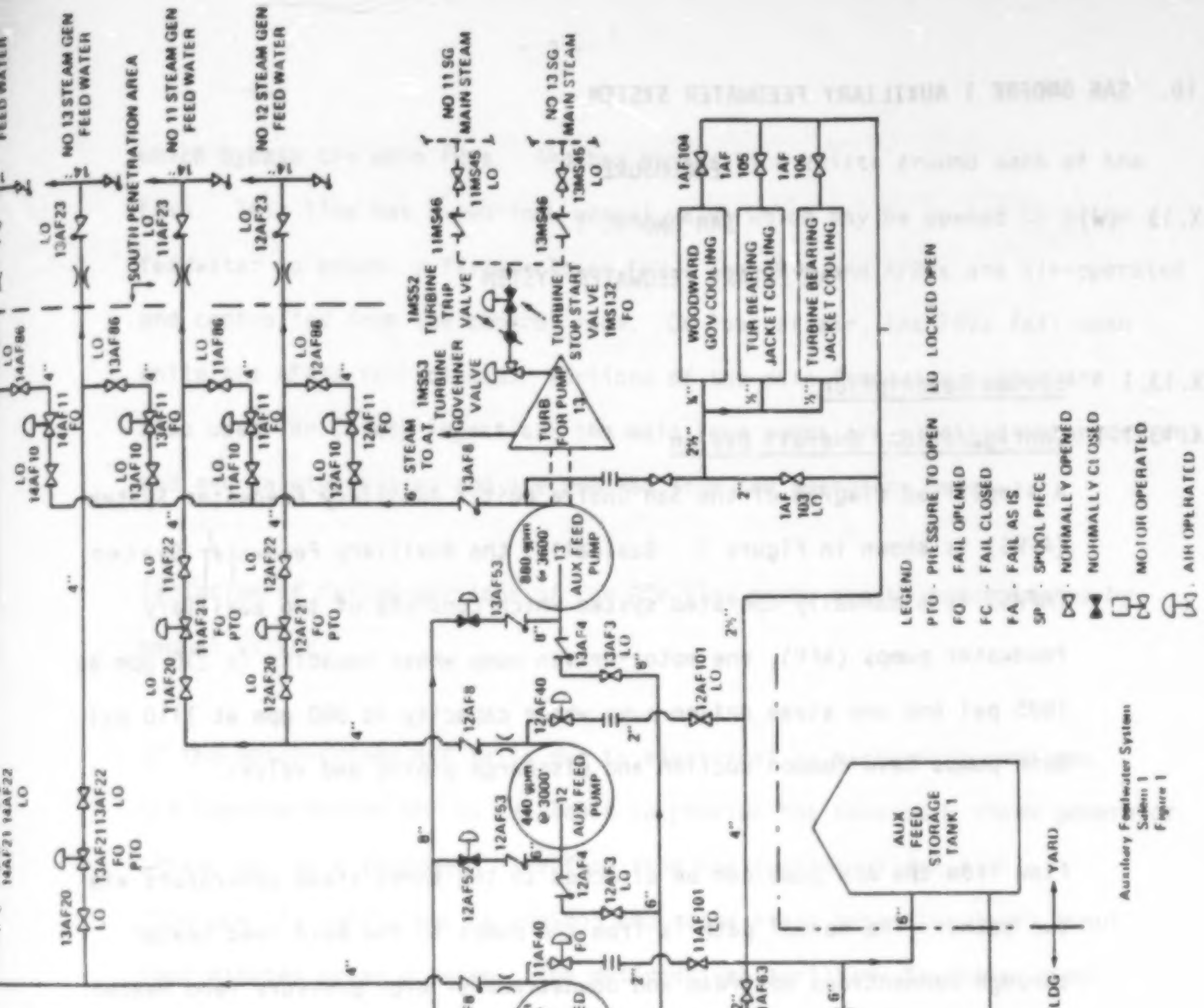
Long-term recommendations for improving the system are as follows:

1. **Recommendation GL-2** - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

   Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

   The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. **Recommendation GL-5** - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.


NO 13 STEAM GEN FEED WATER
SOUTH PENETRATION AREA
NO 11 STEAM GEN FEED WATER
NO 12 STEAM GEN FEED WATER
NO 11 SG MAIN STEAM
NO 13 SG MAIN STEAM
11MS45 LO
13MS45 LO
11MS46
13MS46
1MS52 TURBINE TRIP VALVE
1MS53 TURBINE GOVERNER VALVE
TURBINE STOP START VALVE 1MS132 FO
STEAM TO AT
TURB FOR PUMP 13
880 gpm @ 3600' 13 AUX FEED PUMP
440 gpm @ 3000' 12 AUX FEED PUMP
440 gpm @ 3000' 11 AUX FEED PUMP
WOODWARD GOV COOLING
TUR BEARING JACKET COOLING
TURBINE BEARING JACKET COOLING
1AF104
1AF 105
1AF 106
1AF 103 LO
LO LOCKED OPEN
LEGEND
PTO - PRESSURE TO OPEN
FO - FAIL OPENED
FC - FAIL CLOSED
FA - FAIL AS IS
SP - SPOOL PIECE
NORMALLY OPENED
NORMALLY CLOSED
MOTOR OPERATED
AIR OPERATED
AUX FEED STORAGE TANK 1
YARD
AUX BLDG
DIM WATER MAKEUP
AUX FW STO TANK HX
AF WATER STORAGE TANK HEAT HX WATER CIR PUMP
1AF58
1AF60
1AF61
1AF63
1AF1 LO
11AF3 LO
11AF4
11AF52
11AF53
11AF8
11AF40 FO
11AF101 LO
12AF3 LO
12AF4
12AF52
12AF53
12AF8
12AF40
12AF101 LO
13AF3 LO
13AF4
13AF53
13AF8
11AF20 LO
11AF21 FO PTO
11AF22 LO
12AF20 LO
12AF21 FO PTO
12AF22 LO
13AF20 LO
13AF21 FO PTO
13AF22 LO
14AF21
14AF22 LO
11AF10 LO
12AF10 LO
13AF10 LO
14AF10 LO
11AF11 FO
12AF11 FO
13AF11 FO
14AF11 FO
11AF86 LO
12AF86 LO
13AF86 LO
14AF86 LO
11AF23 LO
12AF23 LO
13AF23 LO
DWST 2
ALT DEM PUMP
FIRE & WATER STORAGE TANK
FIRE & WATER STORAGE TANK
DEMIN WATER STORAGE TANK
DIM WATER T20 PUMPS (3)
TO AUX. FW STORAGE TK MAKEUP
1AF71
1AF72
1AF51
1AF92
2AF 92
SERVICE BLDG
AUX BLDG
TO SALEM 2 AUX FW SYSTEM
SERVICE WATER
2½"
4"
6"
8"
10"
12"
16"
2"
½"

Auxiliary Feedwater System
Salem 1
Figure 1

16. SAN ONOFRE 1 AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

X.13 (W)

SAN ONOFRE 1

AUXILIARY FEEDWATER SYSTEM

X.13.1 System Description

X.13.1.1 Configuration, Overall Design

A simplified diagram of the San Onofre Unit 1 Auxiliary Feedwater System (AFWS) is shown in Figure 1. Basically, the Auxiliary Feedwater System (AFWS) is a manually operated system which consists of two auxiliary feedwater pumps (AFP), one motor-driven pump whose capacity is 235 gpm at 1035 psi and one steam driven pump whose capacity is 300 gpm at 1110 psi. Both pumps have common suction and discharge piping and valves.

Flow from the AFW pump can be directed to the three steam generators via two paths. The normal path is from the pumps to the main feed header through connections upstream and downstream of high pressure feed heater. The second path is the emergency feedwater line which is a four-inch line which can be supplied by either AFP. This line branches into three three-inch lines which join the main feed lines for each of the three steam generators between the main feedwater regulating valves (FRVs) and the main feed line containment penetrations. Normally closed isolation valves in the three-inch lines must be manually opened locally to supply feedwater through the emergency lines. Control of AFW flow through the normal path is by means of air-operated auxiliary feedwater regulating valves(AF which bypass the main FRVs. Another bypass line exists around each of the FRVs. This line has a two-inch manual valve which may be opened to allow feedwater to bypass a failed-closed FRV. The FRVs and AFRVs are air-operated and controlled from the control room. On loss of air, the FRVs fail open while the AFRVs fail closed. Portions of the main feedwater system are also used for safety injection; the main feed pumps are electric motor-driven and are safety-related and are powered from the emergency buses.

Isolation of failed portions of the AFW flow paths can be accomplished by manual valves.

If the motor-driven AFP fails due to electrical or mechanical problems, the turbine-driven AFP is available to provide the necessary steam generator makeup during a shutdown. The flow from the turbine-driven AFP (300 gpm) is sufficient to control and raise steam generator level about four minutes after a scram. The motor-driven auxiliary feedwater pump flow (235 gpm) is sufficient to control and raise steam generator level approximately seven to eight minutes after a reactor scram.

Both AFW pumps receive water via a four-inch line from the condenser make up and reject line, which is connected to the condensate storage tank (CST), with the CST being the primary source of water.

Sources of Water

There are three sources of water for AFW System. The primary source is the Condensate Storage Tank (CST). This tank holds 240,000 gallons of which 15,000 gallons is dedicated to the AFW System. This will last for approximately three hours. All valves to the AFWS are in the normally open position and are manually operated. This tank is not tornado missile protected.

The secondary source of water is the Primary Plant Make-up Tank (PPMT). This tank holds 150,000 gallons of which a maximum of 105,000 gallons is reserved for the AFWS. The Technical Specifications require a total of 105,000 gallons be available either from this tank or the service water reservoir. The PPMT is not tornado missile protected. The licensee estimates that the 105,000 gallons will last approximately 39 hours. The licensee estimates that conservatively 30 minutes may be required to line up the system, since one manual valve must be opened and a primary plant make-up pump (one is normally operating at all times), is used to put water into the CST.

The back-up source and long term cooling is from the Service Water Reservoir through the service water and fire protection systems. This reservoir has a capacity of 3 million gallons of which 105,000 gallons is dedicated as stated above. Portions of these systems, at least the pumps and some of the piping, are not tornado missile protected and would take about 30 minutes to line-up, since manual valves must be opened and a fire hose must be connected to the CST. Complete loss of water sources to the auxiliary feedwater system such as by extensive tornado damage to the CST would disable the AFWS; whereas, tornado damage to the service piping from the service water reservoir affects the availability of the long term supply of water for the AFW system.

X.13.1.2 Components - Design and Classification

Component	Environmental Qualification	Design Classification	Seismic Category
Motor-Driven Pump	Ambient*	Safety Related	B
Steam-Driven Pump	Ambient*	Safety Related	B
AFWS Piping	—	Safety Related	B
Main Feed Piping After main feed pumps	—	Safety Related	
AFWS Valves	Ambient	Safety Related	B
MFWS Valves - After MFW Pumps	Ambient	Safety Related	
Condensate Storage Tank	Ambient	Safety Related	A
Primary Plant Make-Up Tank	Ambient	Non-Safety Related	B
Primary Plant Make-Up Tank Piping System	Ambient	Non-Safety Related	B
Service Water Reservior	Ambient	Safety Related	A
Service Water System at Pumps	Ambient	Safety Related	
Main Steam Piping	—	Safety Related	A

* 40-104°F 100% Humidity

Seismic Catetory

A = Designed for SSE

B = Designed for OBE

C = Non seismic

The system has been reviewed based on documents which are now available to the Staff for postulated breaks in high energy lines including the Main Steam, Main Feed and Auxiliary Feedwater Systems. As a result of the review, we conclude that for a break in the AFW System discharge piping with or without a single active failure, water can be supplied to the steam generator via the main feed pumps and the main feed system assuming these pumps are available and that there is no safety injection signal. A break in the main feed or main steamline outside containment may result in environmental conditions for which components in the main feed and AFWS have not been demonstrated to be operable. A break in the steam line to the turbine-driven AFW pump at the pump may also result in local environmental conditions for which main feed and AFWS components have not been demonstrated to be operable. In this latter case, one train of the main feedwater system would not be affected and would, therefore, remain available to provide feedwater to the steam generators provided there is no safety injection signal. Based on the above, postulated breaks in the main steam and main feed lines may result in local environmental conditions which may disable conventional means to feed the steam generators and result in steam generators boiling dry.

In conjunction with high energy pipe breaks, the licensee states that in accordance with the criteria established by the NRC and previously approved by the NRC for San Onofre Unit 1, the licensee's analysis of pipe breaks outside containment did not postulate breaks in the annulus between the containment and the turbine building. However, in order to protect against the effects of cracks along pipes in this area, the main steam and main feedwater lines were enclosed in metal sleeves. The licensee stated that they consider that steam released from pipe cracks would, for the most part, condense on the sleeves and drip out the ends and any steam which did go out the ends would tend to rise to the open atmosphere. In view of these considerations. The licensee does not consider credible that the environment at the manual auxiliary feedwater valves located approximately 14 feet below the high energy lines would be such as to prevent an operator from opening the valves.

In the feedwater mezzanine area, in order to preclude breaks in the main steam and feedwater lines, an augmented ISI Program has been established. However, breaks were postulated in smaller piping. In addition, cracks were postulated in all piping. To protect cable trays located below the high energy line from jet impingement from the breaks or cracks, the floor grating was replaced with a plate barrier. Although the cable trays penetrate the turbine building wall about 5 feet above the manual auxiliary feedwater valves, the licensee believes that steam is inhibited from passing through these pentrations by the plate barrier. Although steam could pass through penetraions at the elevation of the main steam and feedwater lines, the licensee considers that this steam would tend to rise to the open atmosphere. In the area of the manual auxiliary feedwater valves (about 14 feet below these lines) the licensee believes that the environment would not be so adverse as to prevent an operator from opening the valves. See Section 13.3.3, Long Term Recommendations, and Section 13.3.4, Systematic Evaluation Program Considerations for recommendations relating to high energy pipe breaks.

X.13.1.3 Power Sources

The steam supply for the turbine-driven AFW pump is provided from the main steam header from a connection upstream of the main steam stop valves. The turbine-driven AFW pump is started by local manual startup of the turbine. An air operated valve supplies steam to the turbine and takes power from D.C. Bus 1. On loss of air pressure this valve would fail closed. However, it can be opened manually to control the turbine locally.

The motor-driven pump can be started from the control room, the auxiliary control panel, or with the local operation of its breaker in the 480V switchgear room. The motor receives power from 480V switchgear bus #3. This bus can receive electrical power from both offsite and onsite sources. During a loss of offsite power, emergency diesel generator #1 supplies power to switchgear bus #3 via 4160V bus 1C after the electrical system has been realigned.

The main feedwater regulator motor-operated block valves take their power from the A-C buses and fail in the as-is position on loss of power. These valves can be manually operated locally.

Upon loss of all A-C power, the turbine pump will provide water to the steam generators via manually operated valves. The pump bearings will be cooled by gravity feed from the service water reservior.

X.13.1.4 Instrumentation and Controls

X.13.1.4.1 Controls

All controls for the system are local, manual controls except for the motor driven pump on-off control, the main feedwater regulating valves control and the auxiliary feedwater regulating valve control. These controls are located locally as well as at the remote shutdown panel and the control room. The motor-operated block valves are controlled only locally or in the control room.

X.13.1.4.2 Information Available to the Operator

The following information is available to the operator in the control room.

1. CST, PPMT, and SWR water level alarms
2. CST and PPMT tank level indication
3. Steam Generator Level
4. Steam Generator Low Level Alarms
5. Flow at feed flow control valves
6. Main steam pressure
7. Main Feed Line Pressure
8. Main Feed Flow Control Valve Position Indication
9. Electric AFW pump operation and ammeter

The following information is available at the remote shutdown panel.

1. Steam generator level indication.
2. Electric AFW pump operation

All other information needed by the operator can be found at the local stations.

X.13.1.5 Initiating Signals for Automatic Operation

Since the system is a manually initiated system this section is not applicable, but manual initation is started on loss of main feed pumps and low steam generator level. Subsequent to the staff review of the San Onofre 1 AFW system, the licensee completely revised his emergency operating instructions related to abnormal steam generator water level (including loss of main feed pumps) and steam generator high energy pipe break. These revised procedures identify plant symptoms and provide specific immediate and subsequent action requirements for the control room operator and the dedicated operator stationed at the redundant AFW system manually operated control valves to initiate AFW system operation.

X.13.1.6 Testing

Both Auxiliary Feedwater Pumps are required to be tested bi-weekly, but the licensee states that they are presently being tested weekly in the recirculating mode of operation. The turbine-driven pump is tested every six months in an overspeed condition.

The feed control valves are used continuously for plant operation. All other normally closed valves are not tested except when in use.

The two diesel generators are tested monthly on a staggered bases.

X.13.1.7 Technical Specifications

The Technical Specifications for the plant that are applicable to the Auxiliary Feedwater System are as follows:

TURBINE CYCLE

Operating Status

Applicability: Applies to the operating status of the turbine cycle.

Objective: To define conditions of the turbine cycle necessary to ensure the capability to remove decay heat from the core.

Specification: The reactor shall not be pressurized above 500 psig unless the following conditions are met:

(1) A minimum turbine cycle steam-relieving capability of 5,706,000 lb/hr (except for resting of the main steam safety valves).

(2) Both auxiliary feedwater pumps operable, or the steam-driven auxiliary feedwater pump is continuous operation when the residual decay heat levels are greater than the natural heat losses from the reactor coolant system.

(3) A minimum of 15,000 gallons of water in the condensate storage tank, and an additional 105,000 gallons in the service-water reservoir and/or the primary plant makeup tank.

(4) System piping and valves directly associated with the above components operable.

After criticality is achieved, one auxiliary feedwater pump may be removed from service for maintenance for a period not to exceed 24 consecutive hours.

X.13.2 Reliability Evaluation

X.13.2.1 Dominant Failure Modes

The San Onofre auxiliary feedwater system was analyzed to determine the dominant failure modes under three transient conditions:

a. LOFW with offsite power available

b. LOFW with onsite power available

c. LOFW with only DC power available.

Results of the Anaysis are summarized below.

LOFW with Offsite Power Available

Unavailability of the auxiliary feedwater system is dominated by the following:

a. Operator failure to actuate system upon demand;

b. Failure of the single manual valve in the supply line from the condensate storage tank.

The operator must recognize conditions requiring auxiliary feedwater, start the pumps (electric pump from the control room or turbine pump locally) and locally open the normally closed manual discharge valves. Despite having a dedicated man at the local station, his actions are dependent upon instruction from the control room operator. The availability of the system is, thus, dependent upon the knowledge and actions of the control room operator.

Despite several sources of water, all water is drawn from the condensate storage tank through a single manual valve. Should this valve fail closed, the system will be unavailable.

LOFW with Onsite Power Available

The unavailability of the system is dominated by the same factors as the case discussed above. Postulating loss of one of the two diesel generators does not effect the dominant failure modes.

LOFW with Only DC Power Available

Despite loss of all AC power, the turbine-driven pump train could continue to supply the necessary auxiliary feedwater. Sufficient cooling should be supplied by gravity feed to keep the pump bearings cool.

Short term system unavailability ($\leq$ 30 minutes) is dominated by the potential for maintenance being performed on the turbine driven pump and by the possibility of the discharge block valve inadvertantly being left closed following maintenance on the pump.

Long term unavailability ($\geq$ 30 minutes) is dependent upon assuring that the steam admission valve remains open. This air operated valve will fail closed upon subsequent loss of air which is dependent upon AC power. Manual action is required to open this valve or to provide an air supply after about 30 minutes when local air reservoirs could be depleted.

X.13.2.2 Dependencies

One potential dependency was identified in the analysis. Both pumps of the auxiliary feedwater system are in a common location making them susceptible to any locally adverse conditions such as high energy breaks or fires.

X.13.3 Recommendations for this Plant

The short-term recommendations identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

3.1 Short-Term

1. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendations GL-2 for the longer term resolution of this concern.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered up by the procedures:

   The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and, The case in which the primary water supply is being depleted. The procedures for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes should be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

   The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system vlaves in their normal alignment.

5. Recommendation GS-8 - The licensee should install a system to automatically initiate AFW system flow. This system need not be safety-grade; however in the short-term, it should meet the criteria listed below, which are similar to Item 2.1.7a of NUREG-0578. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-1.

   The design should provide for the automatic initiation of the auxiliary feedwater system flow.

- The automatic initiation signals and circuits should be designed so that a single failure will not results in the loss of auxiliary feedwater system function.
- Testability of the initiating signals and circuits should be a feature of the design.
- The initiating signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits should be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

6. Recommendation - The licensee should propose modifications to Technical Specifications so that manual valves that are normally closed will be tested periodically.

7. Recommendation - The licensee should install valve operators that can be controlled from the control room on all the normally closed manual discharge valves. This will reduce the time delay inherent in

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the present manual set-up as discussed in Section 13.2.1. The AFW system could then be operated from the control room until the system has been fully automated. (See Recommendation 5 above).

8. Recommendation - To reduce dependence on a single flow path from the water sources and increase the quantity of water reserved and readily available for the AFW system, the licensee should connect temporary piping or a fire hose from the Service Water Reservoir/ fire protection system directly to the AFWS pump suction header.

X.13.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or trasnfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shutdown and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578: "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in the Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on the AFW system

train, and there is only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.13.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-1 - Licensees with plants having a manual starting AFW system, should install a system to automatically initiate the AFW system flow. This system and associated automatic initiation signals should be designed and installed to meet safety-grade requirements. Manual AFW system start and control capability should be retained with manual start serving as backup to automatic AFW system initiation.

2. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

   Licensees with plants in which the primary AFW system water supply passes throughvavles in a single flow path, but the alternate AFW system water supplies connected to the AFW system pump suction piping downstream of the above valve(s) should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure. The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

3. Recommendation - GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

4. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure or upgrading the normal source of water to meet seismic Category 1 and tornado protection requirements.

5. Recommendation - The licensee should evaluate the following concerns:

   a) A break in the main feed or main steamline outside containment or a break in the steamline to the turbine driven AFW pump may result in environmental conditions for which the main feed and AFW system components are not qualified.

   b) The San Onofre Unit 1 AFW system design does not meet the high energy line break criteria in SRP 10.4.9 and Branch Technical Position 10-1; namely, that the AFW system should maintain the capability to supply the required AFW flow to the steam generator(s) assuming a pipe break anywhere in the AFW pump discharge lines concurrent with a single active failure.

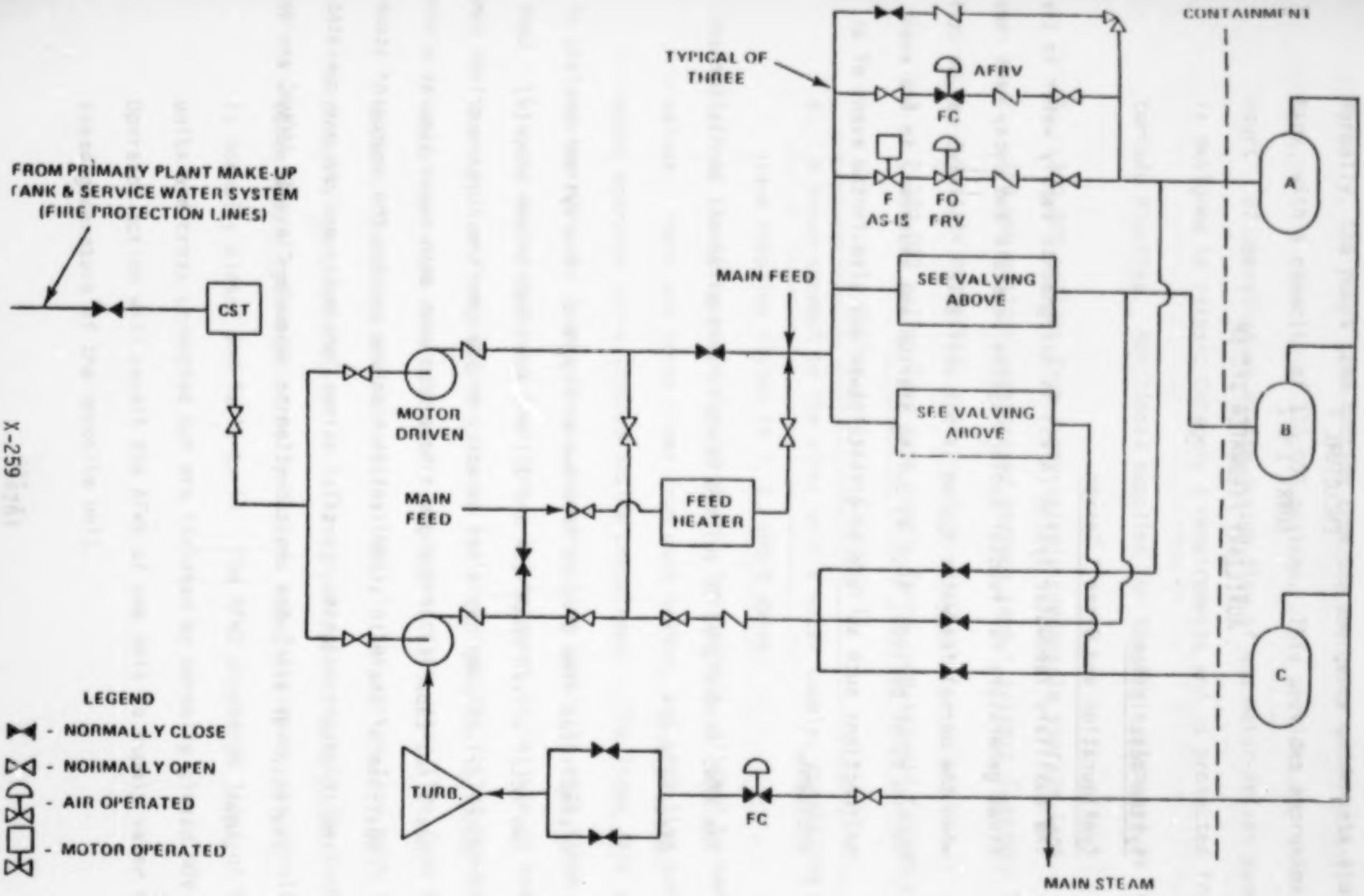
   The licensee should evaluate the postulated pipe breaks stated above and (1) determine any AFW and main feedwater system design changes including environmental qualification, or procedures necessary to detect and isolate the break and direct the required feedwater flow to the steam generator(s) before they boil dry or (2) describe how the plant can be brought to a safe shutdown condition by use of other systems which would be available following such postulated events.

13.3.4 Systematic Evaluation Program Considerations

The following items are still under review by the Systematic Evaluation Program (SEP) and supplement the above long term recommendations:

1. The San onofre Unit 1 plant, including the AFW System, will be reevaluated during the SEP with regard to internally and externally generated missiles, pipe whip and jet impingement including main steam and main feed line breaks inside and outside containment, quality and seismic design requirements, and the effects of earthquakes, tornadoes and floods.

2. The San Onofre Unit 1 AFW System is not automatically initiated and the design does not have capability to automatically terminate AFW flow to a depressurized steam generator and provide flow to the intact steam generator in the event of a main steam or main feed line break. The effect of this design will be assessed in the design basis event evaluations for San Onofre Unit 1.


CONTAINMENT
TYPICAL OF THREE
AFRV
FC
F
AS IS
FO
FRV
FROM PRIMARY PLANT MAKE-UP TANK & SERVICE WATER SYSTEM (FIRE PROTECTION LINES)
CST
MAIN FEED
SEE VALVING ABOVE
SEE VALVING ABOVE
A
B
C
MOTOR DRIVEN
MAIN FEED
FEED HEATER
TURB.
FC
MAIN STEAM
LEGEND
- NORMALLY CLOSE
- NORMALLY OPEN
- AIR OPERATED
- MOTOR OPERATED

Auxiliary Feedwater System
San Onofre 1
Figure 1

17. SURRY 1 AND 2 AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1
SURRY 1 & 2
AUXILIARY FEEDWATER SYSTEM

X.14.1 System Description

X.14.1.1 Configuration and Overall Design

The auxiliary feedwater system (AFWS) is designed to supply water to the steam generators for reactor coolant system sensible and decay heat removal when the normal feedwater system is not available. The AFWS can be utilized during other periods, such as during startup and shutdown, in the event of malfunctions such as loss of offsite power and also in the event of an accident.

The AFWS is designed to seismic Category I requirements and is tornado missile proof.

A simplified flow diagram is shown on Figure 1. The system consists of two motor driven pumps (3A and 3B) and one steam driven pump (2). Each motor driven pump has a net capacity of 350 gpm; the turbine driven pump has a net capacity of 700 gpm. Taps from each main steam line at a point upstream of the main steam isolation valves provide the source of steam to the turbine through two parallel valves, one motor and one air operated. The motor driven pumps are connected to separate Class IE 480VAC emergency buses.

Normally, the pumps take suction from the emergency condensate storage tank, with a capacity of 110,000 gallons. This provides approximately five hours of operation at system design flow of one motor-driven pump. This tank is designed to seismic Category I requirements and is protected from tornado missiles. Additional supplies for the AFW system are as follows:

1. 300,000 gallons from a non-seismic condensate storage tank.
2. 110,000 gallons underground storage tank - seismic Category I and missile protected.
3. 2-300,000 gallon Fire Main Supply Tanks (non-seismic Category I), Fire Main Supply Piping (seismic Category I).
4. A cross-connect to the other unit's water supply, consisting of the same supplies listed in 1, 2, and 3 above.

Each of the three pumps discharge into two headers, aligned by manual valves. There are three lines from each header, and each line contains a motor operated valve located inside containment. The lines join downstream of the MOVs and form a common discharge line supplying each steam generator via the associated main feed line. In the event of failure of one header, the supplies from the pumps may be isolated from the failed header by manual operated valves to assure steam generator water flow from the other header. The motor operated valves (MOV) in the system flow path are normally open, and fail as-is. The air operated valve in the turbine steam supply system is normally closed, and fails open; the parallel MOV is normally closed and fails as-is. The AFWS discharge lines of both units are cross connected but are isolated by normally closed MOV valves. Operator action will permit the AFWS of one unit to supply water to the steam generators of the opposite unit.

X.14.1.2 System Design Classification

The turbine pump train and motor pump trains (pumps, valves, motors, piping) are seismic Category I and tornado missile proof (Class 1E for electrical equipment).

X.14.1.3 Power Sources

The motor driven pumps and valves are supplied from the Class 1E A-C emergency buses; 3A from Emergency bus 1H, 3B from emergency bus 1J. The air operated turbine pump steam admission valve is D.C. solenoid operated and fails open. The parallel MOV is powered by Class 1E A-C power and fails as is.

X.14.1.4 Instrumentation and Control

The instrumentation and control power supplies are from the 120 VAC vital bus system. There are four vital buses, two supplied from inverters connected to the emergency DC power supplies and two regulated power supplies connected to the AC emergency buses.

X.14.1.4.1 Controls

Steam generator level is controlled manually from either the main control room or the auxiliary shutdown panel by operating the appropriate MOV in the AFW line. The valves are motor operated, are normally open and fail as-is on loss of power. Class 1E instrumentation is provided (level and flow indications).

Information Available to the Operator

Except as noted, the following indications are available at both operating stations:

1. MOV position indication
2. Air operated valves position indication, turbine control system
3. Auxiliary feedwater flow (Control Room only)
4. SG level-wide range
5. Auxiliary feed pump amperage (Control Room only)
6. Breaker (motor driven pump) position
7. Condensate (110,000 gal) tank level (Control Room only)

Initiating signals for Automatic Operation

1. The following signals start the motor driven pump motors:
    a. Low-Low level from any steam generator
    b. Undervoltage on transfer buses D & F
    c. Safety injection
    d. Trip of both main feed pumps
    e. Manual

2. The following signals open the steam control valves starting the steam turbine:
    a. Undervoltage on the Station Service bus (2 out of 3 logic)
    b. Low-Low steam generator level-2 out of 3 steam generators
    c. Manual

X.14.1.5 Testing

The systems are tested every 30 days in accordance with technical specification requirements. In addition to the periodic tests, operational tests are performed in accordance with surveillance tests following maintenance on a particular system or component. The instrumentation systems are tested periodically, per shift, every 30 days and every 18 months in accordance with technical specification requirements.

X.14.1.6 Tech Specs

The limiting condition of operation (LCO) permits plant operation if two of the three auxiliary feedwater pumps are operable. This could result in unrestricted plant operation if any of the three (including the steam driven pump train) remains inoperable.

X.14.2 Reliability Evaluation Results

X.14.2.1 Dominant Failure Modes

Successful delivery of feedwater is considered to be the flow of at least 350 gpm to one (or more) of the three steam generators, for the transients considered here.

Failure modes of the AFWS were assessed for three types of initiating transients. The dominant failure modes for each transient type are discusse below.

Loss of MFW with offsite power available

The reliability analysis of the Surry AFWS based on this initiating transient did not identify any single failures or double failures which would fail the entire AFWS. Consideration was given to combinations of three failures such as a combination of one pump out for maintenance, coupled with hardware failures in the other two lines. However, the dominant failure mode was assessed to be a common cause failure resulting from the failure to reopen all the manual pump discharge valves following test, coupled with the failure to either reopen at least one valve within approximately one-half hour after a demand on the AFWS, or to draw feedwater from the other unit's AFWS through the unit intertie connections.

Loss of MFW with only onsite AC power available

The response of the Surry AFWS to this transient should not be significantly different from that for the case discussed above. As such, it is again concluded that the dominant failure mode is the human error of failing to reopen the manual pump discharge valves after test, coupled with the failure to reopen one valve or to realign the other unit's AFWS within 30 minutes after an AFWS demand.

Loss of MFW with only DC power available

In this transient neither onsite or offsite AC power is available; thus the AFWS is reduced to the one steam-driven pump train. Failure of this train can occur in a number of ways. The results of this examination indicate that the dominant mode of failure is that the steam-driven train is out of service due to maintenance. The current Surry Technical Specifications and LCOs permit the outage of one AFWS pump indefinitely, so that the possibility of the steam train being out of service in a station blackout incident could be high. Revision of the Surry Technical Specifications and LCOs to normally require the operability of all three trains except for limited maintenance outages (as in the standard Tech Specs) would improve the reliability of the AFWS substantially for this transient.

X.14.2.2 Potential Dependencies

The potential for a common-cause failure of the AFWS due to human error is, as discussed above, the most significant dependency found in this analysis. A second potential common-cause failure due to commonalities of equipment location was also noted; however, since the unit intertie system was installed specifically to alleviate this possible problem, this does not appear to be a significant concern.

X.14.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.14.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumention can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.

- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

- Testability of the initiation signals and circuits shall be a feature of the design.

- The initiation signals and circuits should be powered from the emergency buses.

- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

6. Recommendation - Procedures should be established to lock open and periodically verify open position of all manual AFWS valves inside containment.

7. Recommendation-The licensee should require staggering of the periodic pump train tests (e.g., one train at North Anna is tested every 10 days rather than all three trains tested at once on a monthly basis). This reduces the potential for inadvertently leaving closed the discharge valves of all trains after test.

8. Recommendation - Emergency procedures should be available to the operators for operating the AFWS of one unit such that it is supplying water to the steam generator(s) of the opposite unit in the event that such an operating mode should be necessary.

X.14.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and

run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/ bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

> "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.
>
> The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technnical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room

be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

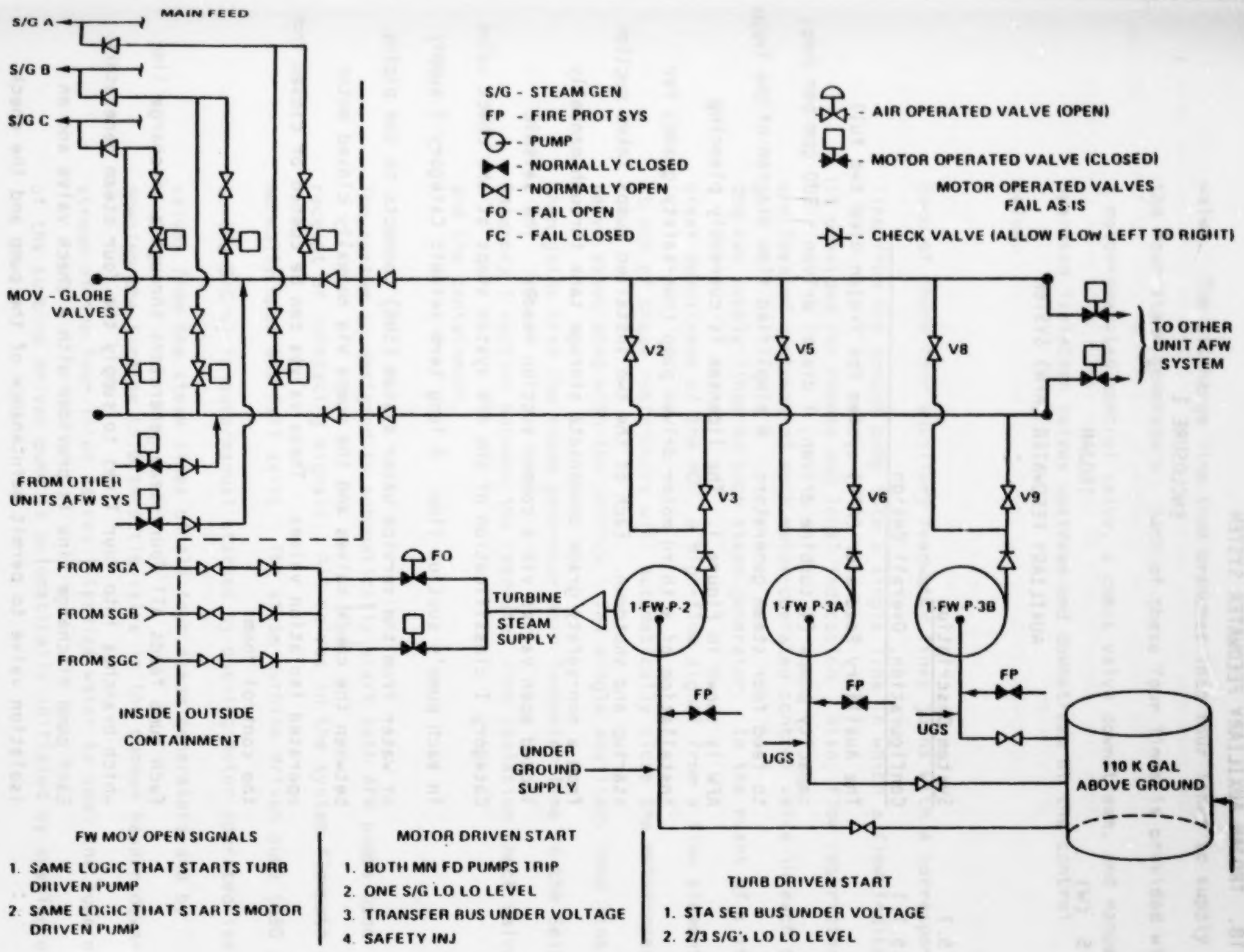
X-14.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.
2. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.
3. Recommendation - The AFWS flow control valves for both the motor and turbine pump trains are AC powered, normally open, fail as-is motor operated valves which are located inside containment. Also, manual, normally open valves are located inside containment. The AFWS design should be reevaluated, including the possibility of relocating the valves outside containment, assuming an accident inside containment which necessitates AFWS operation and which creates a containment environment (humidity, radiation) that precludes access to the valves. The reevaluation should consider the following:

a. A possible common mode failure (environmentally induced) causing spurious closure or failure of the MOV's in a throttled position.

b. An AFWS line break downstream of the MOV's and failure of the MOV's to operate.


S/G A
MAIN FEED
S/G B
S/G C
S/G - STEAM GEN
FP - FIRE PROT SYS
- PUMP
- NORMALLY CLOSED
- NORMALLY OPEN
FO - FAIL OPEN
FC - FAIL CLOSED
- AIR OPERATED VALVE (OPEN)
- MOTOR OPERATED VALVE (CLOSED)
MOTOR OPERATED VALVES FAIL AS IS
- CHECK VALVE (ALLOW FLOW LEFT TO RIGHT)
MOV - GLOBE VALVES
V2
V5
V8
TO OTHER UNIT AFW SYSTEM
FROM OTHER UNITS AFW SYS
V3
V6
V9
FO
FROM SGA
FROM SGB
FROM SGC
TURBINE STEAM SUPPLY
1-FW-P-2
1-FW-P-3A
1-FW-P-3B
FP
FP
FP
INSIDE
OUTSIDE
CONTAINMENT
UGS
UGS
UNDER GROUND SUPPLY
110 K GAL ABOVE GROUND
FW MOV OPEN SIGNALS
1. SAME LOGIC THAT STARTS TURB DRIVEN PUMP
2. SAME LOGIC THAT STARTS MOTOR DRIVEN PUMP
MOTOR DRIVEN START
1. BOTH MN FD PUMPS TRIP
2. ONE S/G LO LO LEVEL
3. TRANSFER BUS UNDER VOLTAGE
4. SAFETY INJ
TURB DRIVEN START
1. STA SER BUS UNDER VOLTAGE
2. 2/3 S/G's LO LO LEVEL

Auxiliary Feedwater System
Surry Nuclear Station
Figure 1

18. TROJAN AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

X.15 (W)

TROJAN
AUXILIARY FEEDWATER (AFW) SYSTEM

X.15.1 System Description

X.15.1.1 Configuration, Overall Design

The Auxiliary Feedwater (AFW) system for Trojan uses two full capacity pumps (1 turbine driven, 1 diesel driven - 880 gpm per pump) to feed four steam generators. A simplified flow diagram of the Troja AFW is shown in Figure 1. The licensee is currently planning installation of a third motor driven pump (non-safety grade) for startup and shutdown. Each of the two installed pumps takes suction from a non-safety grade condensate storage tank through manually locked open valves via a common suction header. The seismic Category I classification of the AFW system stops at the check valve in each pump's suction line. A long term seismic Category I supply of water from the service water system (SWS) connects to the piping between the check valves and the pumps via normally closed motor operated isolation valves. These valves can be opened or closed from the control room.

Each pump feeds all four steam generators through a discharge line which branches into four lines to supply the four steam generators. Each pump discharge line is provided with a check valve and an isolation valve to permit maintenance of the pump and the check valve. The discharge line then branches into four lines to supply the four steam generators. Each of these four lines is provided with a motor-operated control valve, a check valve downstream, and manually-operated isolation valves upstream and downstream of the control valve.

Each of these four auxiliary feedwater lines joins with a corresponding line from the second pump into a single line in which a flow indicator is provided for remote and local indication. Also, flow restrictors are located upstream of each motor-operated control valve in each of the two supply lines to each steam generator. In the event of a pipe break downstream of the MOV, a high-flow signal from a flow element at one of these restrictors will automatically close the motor-operated valve associated with the orifice. The single auxiliary feed line then joins with the steam generator main feedwater line in the Seismic Category I section between the feedwater line isolation check valve and the containment.

The system is designed to automatically start both AFW pumps upon receipt of initiating signals. All valves in the system flowpath are normally open and fail as-is. The steam turbine driven pump (880 gpm to the S/G's) is auto/manual started by opening motor operated isolation valves from the steam lines of all four steam generators and by opening the turbine trip throttle valve in the common header downstream of the four inlet valves. Service water to cool the lube oil of the turbine driven pump is automatically initiated by opening of a

MOV from the service water system whenever the turbine pump gets a signal to start. The licensee is presently revising this design such that the lube oil will be cooled by the discharge of the turbine driven pump.

The diesel driven pump has its own starting battery which automatically starts the diesel. The initiating signals also start a reduction gear lube oil priming pump, and open an MOV to supply service water for jacket cooling and lube oil cooling. The diesel has a day tank with a 500 gallon capacity good for 10 hours of diesel operation. Automatic transfer of oil to the day tank from the emergency diesel fuel oil transfer system is controlled by day tank level.

Both the diesel and turbine driven pumps use governors that control the speed to automatically maintain a set pressure differential between the pump discharge and the steam generators. This pressure differential can be selected by the control room operator to help control steam generator level.

X.15.1.2 Components, Design Classification

The condensate storage tank and the piping from the condensate storage tank to the check valve in each pump's suction line are non-safety grade (non-seismic). The recirculation lines from the pump discharge to the condensate storage tank are also non-safety grade. The pumps, piping, valves, and valve actuators for the rest of the AFW system are seismic Category I.

The controls, instrumentation and power supplies for the operation of the auxiliary feedwater system are seismic Category I, Class IE. However, the actual indicators in the control room are not designed to meet seismic Category I requirements.

The ventilation supply fans, diesel fuel oil and lube oil system, service water cooling supply and water source supply are designed to seismic Category I requirements.

X.15.1.3 Power Sources

The turbine driven pump (train A) is associated with the train A electrical buses,including the train A ESF channels for automatic operation.

The diesel driven pump is supplied by train B in the same manner as the turbine driven pump.

Neither train is independent of AC power. The steam inlet valves to the turbine driven pump are operated from the train A vital 480 volt AC bus. Manual operation of these valves is required to open these MOV's upon loss of the train A vital bus,since they fail as-is and are normally closed. These valves are normally closed to protect against the effects of a steam line break in the supply line downstream of these valves.

The diesel driven pump may start and operate without AC power but due to lack of cooling water to the jacket and lube oil, and due to lack of ventilation, operation of the diesel could not be sustained. The licensee estimates that this diesel will trip on over-temperature in 5-10 minutes.

The vital DC buses are used to supply control power to the speed governors for both pumps and for operation of the turbine trip throttle valve to the turbine driven pump.

X.15.1.4 Instrumentation and Controls

X.15.1.4.1 Controls

All controls for normal operation for the AFW system are Class 1E and operated from the control room. These include steam inlet valves, the trip throttle, the steam control valve in the steam line to the turbine, the AFW flow control valves (2 to each S/G), and pump start and stop.

Steam Generator level is controlled from the control room by controlling pump speed and opening/closing of the AFW flow control valves to the steam generator. Steam generator level transmitters and instrumentation circuits are safety grade but the level indicators on the control panel are not.

X.15.1.4.2 Information Available to the Operator

I. Alarms

1. Hi/Lo Steam Generator Level
2. Lo Condensate Storage Tank Level
3. Low AFW Pump Suction Press-each pump
4. Local-Control Override for each pump
5. Hi Conductivity AFW

II. Indication

1. Steam Generator Level
2. Condensate Storage Tank Level
3. Valve Position Indication for all MOV's, including SWS
4. Steam Pressure at Turbine
5. Suction Press each pump
6. Discharge Press each pump
7. Auto/Man Light for Pump Control
8. AFW Flow to each Generator (not for each pump)
9. Differential Pressure - Pump discharge and steam generator

X.15.1.4.3 Initiating Signals for Auto Operation

Both Pumps 1. Safety Injection Signal

2. Lo-Lo Level in any steam generator (2 out of 3 detectors)

3. Loss of Both main Feed Pumps
4. Loss of Offsite Power - Sensed on Vital Bus

X.15.1.5 Testing

Pumps and motor operated valves are tested monthly. All the MOV's, including the service water system supply isolation valves, are cycled during their monthly test. The pumps are tested by closing the flow control valve for the pump, starting the pump, and checking pressure and recirculation flow. All testing is done from the control room.

Every 18 months, a flow verification test from the condensate storage tank to the steam generators is performed. Also every 18 months, automatic start of the AFW pumps from the auto-start logic is tested. Every 18 months, a routine instrumentation and controls calibration check is performed.

X.15.1.6 Technical Specifications

1. At least two independent steam generator auxiliary feedwater pumps and associated flow paths shall be operable with:
   a. One feedwater pump capable of being powered by an operable diesel with ≥450 gallons of fuel in its day tank, and
   b. One feedwater pump capable of being powered from an operable steam supply.

2. With one AFW pump inoperable, restore the inoperable pump to operable status within 72 hours or be in hot shutdown within the next 12 hours.

X.15.2 Reliability Evaluation

X.15.2.1 Dominant Failure Modes

LOFW with Offsite Power Available

The dominant failure contributors include a combination of human errors associated with the two water sources (manual valve left closed and failure to take corrective actions). Other dominant contributors are combinations of hardware failures associated with each pump train, test and maintenance outages, and a human error resulting in manual valves left closed in the pump discharge lines, undetected by control room indication or by pump test indications.

LOFW with Loss of Offsite Power with Onsite AC Power Available

The dominant failure contributors are the same as for the non-LOP transient with the addition of a single emergency AC train failure in combination with other failures in the other pump train.

LOFW with Loss of All AC, DC Available

Under present design, assuming completion of the modification to provide bearing cooling water from the AFW turbine pump line, the dominant failure contributors are single failures. They include the human failure to open a condensate storage tank manual valve, the hardware, test and maintenance and human error contributors associated with the turbine train-human failure to open the AC steam inlet MOV's by hand and AC power dependence for cooling the diesel driven pump.

X.15.2.2 Interdependencies

The principal noted dependency is the design for AC cooling of the diesel driven pump and for operation of the turbine steam inlet valves.

X.15.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading c Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this se tion involve system design evaluations and/or modifications to improve A system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

15.3.1 Short Term

1. Recommendation GS-2 - The licensee should lock open single valves multiple valves in series in the AFW system pump suction piping an lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be perfo to verify that these valves are locked and in the open position.

These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions.

Since the water for cooling of the lube oil for the turbine-driven pump bearing may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures and, if necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer term resolution of this concern.)

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   Proceduresshould be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

   The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5. Recommendation GS-7 - The licensee should verify that the automatic start AFW signals and associated circuitry are safety grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety grade requirements as indicated in Recommendation GL-5.

   The design should provide for the automatic initiation of the auxiliary feedwater system flow.

   The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

   Testability of the initiation signals and circuits shall be a feature of the design.

   The initiation signals and circuits should be powered from the emergency buses.

   Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

   The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.15.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.
2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test

acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   > "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.
   >
   > The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.10.3.3 Long Term

Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

   Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s) should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

   The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation - GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

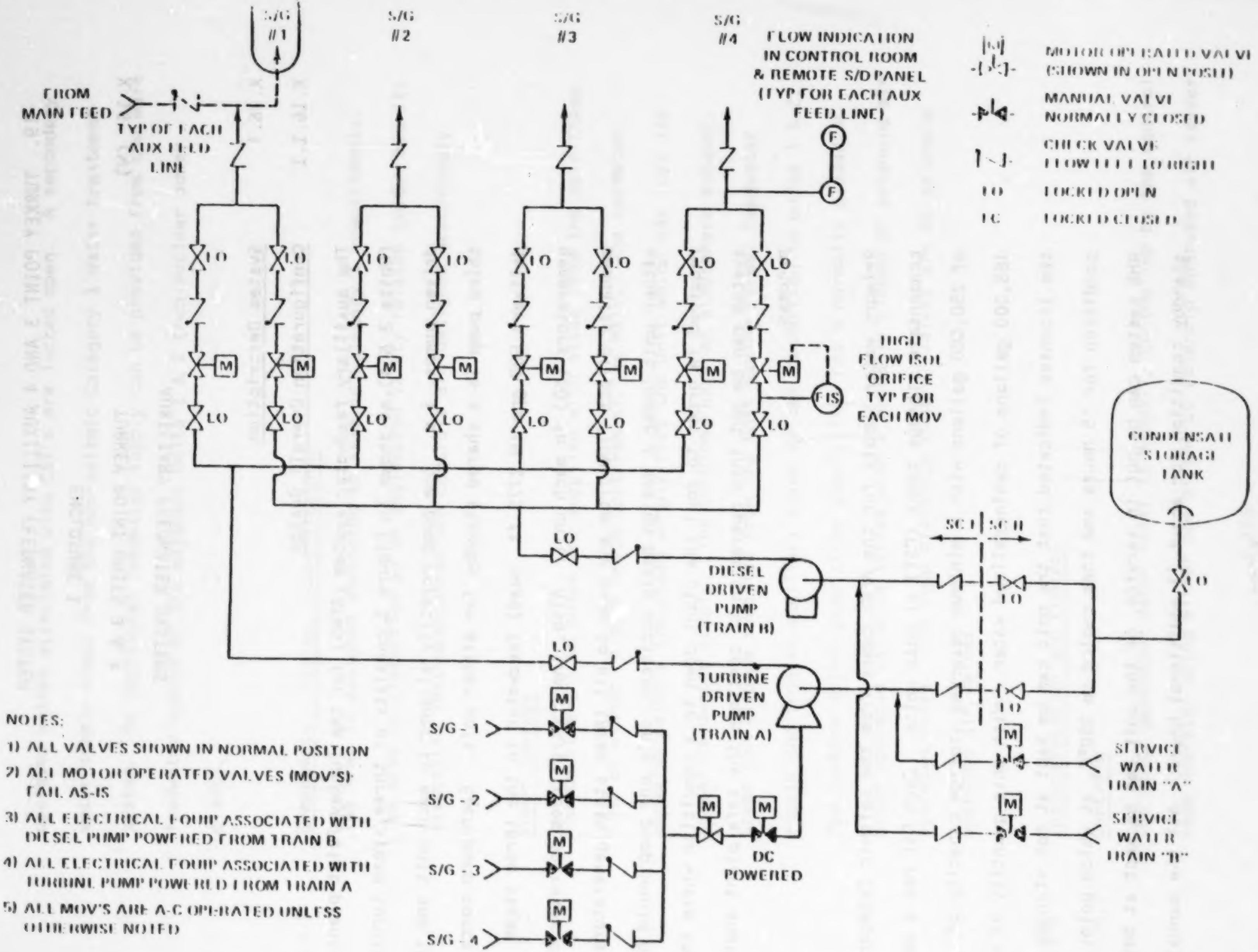
3. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of

their AFW systems to determine if automatic protection of the pump is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available for the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

4. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

5. Recommendation - A motor driven pump is currently being installed or is planned to be installed by the licensee. Present plans are for a non-safety grade motor driven pump system. Based on past experience of the problems associated with the speed control (overspeed trips) of both the diesel and turbine driven pumps and other Licensee Event Reports on the Trojan AFW system, the licensee should further review the proposed installation to determine if the motor driven pump should be safety grade and automatically actuated by the AFW automatic start logic.

6. Recommendation - A pipe break in certain locations of the turbine driven auxiliary feedwater pump discharge piping may affect both AFW trains, since portions of this piping pass through the diesel driven pump room. The motor driven pump to be installed should be located such that a break in the AFW system (not associated with the motor driven pump train) could not affect the motor drive pump. Also the licensee should 1) install the motor pump with appropriate valves in the pump discharge line connections to meet the high energy line break criteria in SRP 10.4.9 and Branch Technical Position 10-1; namely, the AFWS should maintain the capability to supply the required AFW flow to the steam generator(s) assuming a pipe break anywhere in the AFW pump discharge lines plus a single active failure or 2) describe how the plant can be brought to a safe shutdown condition by use of other available systems following such a postulated event.



S/G #1
S/G #2
S/G #3
S/G #4
FROM MAIN FEED
TYP OF EACH AUX FEED LINE
FLOW INDICATION IN CONTROL ROOM & REMOTE S/D PANEL (TYP FOR EACH AUX FEED LINE)
F
F
MOTOR OPERATED VALVE (SHOWN IN OPEN POSIT)
MANUAL VALVE NORMALLY CLOSED
CHECK VALVE FLOW LEFT TO RIGHT
LO LOCKED OPEN
LC LOCKED CLOSED
LO
M
HIGH FLOW ISOL ORIFICE TYP FOR EACH MOV
FIS
CONDENSATE STORAGE TANK
SC I
SC II
DIESEL DRIVEN PUMP (TRAIN B)
TURBINE DRIVEN PUMP (TRAIN A)
S/G - 1
S/G - 2
S/G - 3
S/G - 4
DC POWERED
SERVICE WATER TRAIN "A"
SERVICE WATER TRAIN "B"
NOTES:
1) ALL VALVES SHOWN IN NORMAL POSITION
2) ALL MOTOR OPERATED VALVES (MOV'S) FAIL AS-IS
3) ALL ELECTRICAL EQUIP ASSOCIATED WITH DIESEL PUMP POWERED FROM TRAIN B
4) ALL ELECTRICAL EQUIP ASSOCIATED WITH TURBINE PUMP POWERED FROM TRAIN A
5) ALL MOV'S ARE A-C OPERATED UNLESS OTHERWISE NOTED

Auxiliary Feedwater System
[illegible] Nuclear Plant

19. TURKEY POINT 3 AND 4 AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

X.16 (W) TURKEY POINT UNITS 3 & 4 AUXILIARY FEEDWATER SYSTEMS

X.16.1 System Description

X.16.1.1 Configuration-Overall Design

The auxiliary feedwater system (AFWS) for the Turkey Point plant (Units 3 & 4),as shown in Figure 1, consists of three steam turbine driven pumps, i.e., one pump normally aligned to each unit and the third pump is a shared standby for either unit. Each pump normally delivers 600 gpm (@ 2775 ft. head) feedwater to the three steam generators (SG) in each unit. Also, the control room operator can manually direct flow from any pump to all three steam generators of either unit.Under a design basis accident, only one pump would be required in order to cool the plant down to a condition where the RHR system can be put into operation to continue the safe plant shutdown process.

Primary water supply for the AFWS comes from the seismic Category I condensate storage tanks (CST) of both units. Each CST has a capacity of 250,000 gallons with a minimum reserved storage capacity of 185,000 gallons of demineralized water. With this quantity of water, the licensee indicated that the unit can be kept at hot standby condition for 15 hours and then cooled to 350°F, at which point the RHR system can be put in service, or the unit can be kept at hot standby condition for about 23 additional hours. All the manually operated valves associated with CST's are locked open. A secondary water supply comes from the non-seismic Category 1 water treatment system. An additional feedwater supply can be provided from the main feedwater system of the adjacent Units 1 & 2 (non-nuclear power plant).

X.16.1.2 Components - Design, Classification

The AFWS is designed according to seismic Category I requirements. The AFWS is classified as an engineered safety related system and its associated instrumentation and controls are designed accordingly.

X.16.1.3 Power Sources

The turbine driven pumps are supplied with steam from the main steam line of either or both units upstream of the MSIV. The operator normally selects the steam supply from the Unit which has lost its normal feedwater supply. The turbines have an atmosphere exhaust. Steam can also be supplied from the Unit having normal feedwater supply and from an auxiliary steam system connection to Units 1 & 2. The turbine driven pump steam supply line has a normally closed AC motor operated valve in series with a normally closed DC solenoid air operated valve. The pump discharge control valves are DC solenoid operated air valves.

X.16.1.4 Instrumentation and Control

X.16.1.4.1 Controls

The steam generator water level is manually controlled by the control room operator using either one of the DC solenoid operated air valves.

Local manual operation of these valves can be performed on loss of compressed air. The AFW pump feedwater discharge rate is always greater than the turbine steam consumption when the steam pressure is higher than 120 psig. When the steam pressure is reduced to 120 psig, the RHR system is started and the AFW pumps are shut down.

X.16.1.4.2 Information Available to Operator

Low water level in the condensate storage tank will alarm and annunciate in the main control room. In addition, AFW flow indication, SG water level, and control valve position indication are provided in the control room.

X.16.1.4.3 Initiating Signals for Automatic Operation

All three AFW pumps will automatically start by any of the following signals from either Unit:

(a) safety injection

(b) low-low water level in any of the three steam generators

(c) loss of voltage on both 4160V buses

(d) loss of both main feedwater pumps.

Any one of these signals will also automatically open the normally closed motor operated and air operated valves in series which isolate the main steam line from the steam supply header of each AFW pump turbine. Air to operate the AFW control valves to the steam generators is supplied when the steam supply valves commence opening. The AFWS can also be started manually in the control room or from the local station.

X.16.1.5 Testing

The Turkey Point Units 3 and 4 Technical Specifications require the following testing of the auxiliary feedwater system.

1) Monthly test of each auxiliary feedwater pump to run for 15 minutes and verify a flow rate of 600 gpm to the steam generators.
2) Tests of auxiliary feedwater discharge valves during the monthly pump tests.
3) Tests of steam supply and turbine pressure valves during monthly pump tests.

These tests are designed to verify the operability of the auxiliary feedwater system and its ability to respond properly when required.[1]

X.16.1.6 Technical Specifications

The Turkey Point Units 3 and 4 Technical Specifications provide for the following limiting conditions for operation with respect to the Auxiliary Feedwater System:

1) Two out of three AFWS pumps must be operable for single nuclear unit operation.
2) Three out of three AFWS pumps must be operable for dual nuclear unit operation.

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[1] The licensee advised that the type of periodic (monthly) testing performed for the AFWS includes full flow path discharge to the SG's, i.e., a single actuation of AFWS and delivery to SG's while power is being produced.

3) The condensate storage tank must contain a minimum of 185,000 gallons of water.

4) System piping, interlocks and valves must be operable.

If any of the above conditions cannot be met within 48 hours, the reactor must be shut down and the reactor coolant temperature must be reduced to less than 350°F.

X.16.2 Reliability Evaluation

X.16.2.1 Dominant Failure Modes

The AFWS simplified flow diagram for Turkey Point Unit 3 is illustrated in Figure 1. This AFWS design reflects a redundant, highly shared, system between Units 3 and 4. Operation of any one of the three steam turbine driven pumps would be expected to result in successful decay heat removal from either Units 3 or 4. Accordingly, the success criterion selected for this reliability evaluation was: Failure of AFWS is insufficient AFWS flow from one AFWS pump to 2 of 3 steam generators in one unit.

The following failure modes were found to dominate the demand unavailability of the Turkey Point AFWS.

LOFW with Offsite AC Available

The Turkey Point AFWS was found to be highly redundant in that there was no obvious single faults (active components, manual valves or human errors) identified that dominate the availability of the AFWS.

The periodic testing practice followed involves full flow path testing to the steam generators. This type of testing is of quality in that it yields an advantage on detectability of valves that might be mispositioned through human errors. Also, the AFWS manual valves are locked open and this practice further reduces the chance of inadvertent closure through human error.

Several unlikely common mode vulnerabilities were identified that might serve to limit the availability of the highly redundant Turkey Point AFWS; their ultimate impact should be further considered in a longer time assessment. These were:

a) The possible common sharing of the lube oil cooling by the service (city) water system which is DC powered.

b) The potential for common disabling of Unit #3 and/or #4 AFWS by a single failure of the connecting piping between the headers in the AFWS pump discharge and steam supply paths.

LOFW with Only Onsite AC Available

The impact of shared emergency diesel generators (EDG) and their contribution to the unavailability of the Turkey Point AFWS were estimated to be very small. The steam admission valves to the turbine pumps are AC operated,but either of the two EDG's operating would suffice to operate at least one or more of the three AC valves in each header in Unit 3 and 4. Further, the human can serve as backup to open these valves if for some reason, the AC or DC valves in either Unit 3 or 4 steam admission header failed to operate electrically.

The dominant faults appear to remain similar to those discussed for the preceeding LOFW transient event.

LOFW with Only DC Available

As noted above the steam admission valves are AC operated in Turkey Point Units #3 and #4. The dominant fault contribution for this event was assessed to be failure of the human to open at least one of the steam admission valves by local manual action. The licensee estimated that such actions could be accomplished within about 10 minutes.

X.16.2.2 Principle Dependencies Identified

One dependency identified was the AC dependency for the steam admission valves that,for the event including complete loss of AC,would require local manual action to initiate the AFWS.

Several additional dependencies were identified that should be considered further,but on a longer term consideration as to their ultimate impact on the AFWS. These were (a) the potential for common lubrication cooling faults in the service (city) water system and (b) the potential for common disabling of the AFWS due to breaks in the single line in the AFWS discharge headers and in the steam supply headers to all turbine driven AFW pump turbines.

X.16.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.16.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time period that one AFW system pump and its associated flow train and essential instrumentation can be inoperable.

   The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are

locked and in the open position. These inspections should be proposed for incorporated into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

3. Recommmendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

4. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation of flow control is

required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling the lube oil for the turbine-driven pump may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations for manual initiation and control of the AFW system should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer term resolution of this concern).

5. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the intiation signals and circuits shall be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and

should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultanous and/or sequential) of the loads to the emergency buses.

The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to intiate the AFW system from the control room.

X.16.3.2 Additional Short Term Recommendations

The following additional short term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins & Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs in W- and C-E-designed operating plants. They have not been examined for specific applicability to ths facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator actions, assuming that the largest capacity AFW pump is operating.

2. **Recommendation** - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain with design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperatures, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. **Recommendation** - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

> "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.
>
> The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. **Recommendation** - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for

operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.16.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

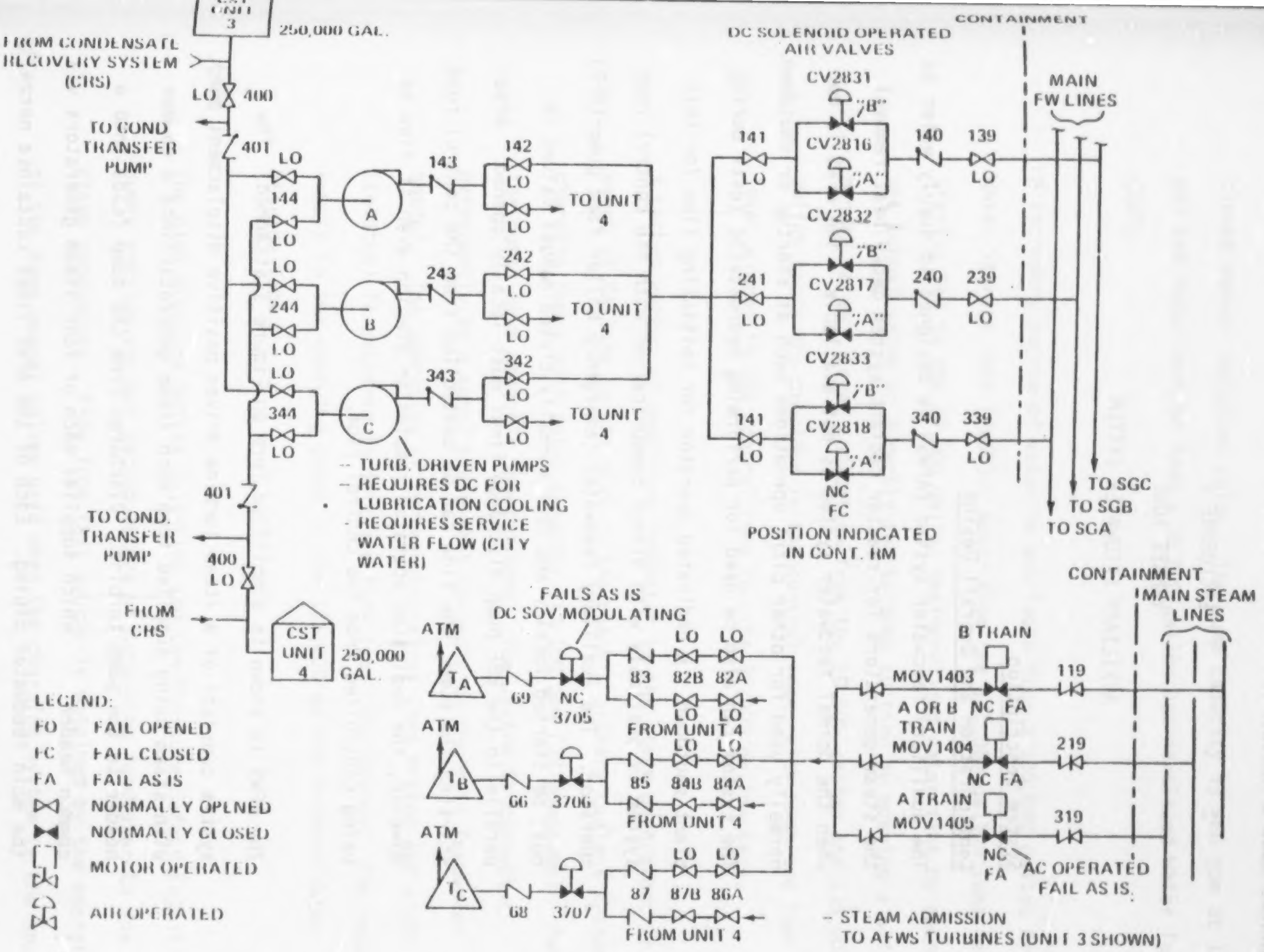
1. Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

2. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

3. Recommendation - The AFW pump discharge lines and turbine driven AFW pump steam supply lines for each unit combine into single lines through which all water and steam respectively from either unit must flow. A pipe break in either of these single flow paths would cause loss of the capability to provide AFW flow to all the steam generators of one unit. The licensee should evaluate the consequences of a

postulated pipe break in these sections of the AFW discharge or steam supply, assuming a concurrent single active failure and 1) determine any AFW system modifications or procedures necessary to detect and isolate the break, and direct the required AFW flow to the steam generators before they boil dry or 2) describe how the plant can be brought to a safe shutdown condition by use of other available systems following such a postulated pipe break.

4. Recommendation - The lube oil cooling of the three turbine driven AFW pumps is provided from a common source; namely the service (city) water system. The licensee should evaluate this cooling water system to determine if there are potential common mode (electrical or mechanical) failures that could disable the lube oil cooling for all three turbine driven pumps. The licensee should provide the results of the evaluation and 1) indicate any system modifications or procedures necessary to prevent a common mode failure of the lube oil cooling system or 2) provide information that demonstrates that the turbine driven AFW pumps can operate for at least two hours without lube oil cooling water and independent of AC power.


CST UNIT 3
250,000 GAL.
FROM CONDENSATE RECOVERY SYSTEM (CRS)
LO 400
TO COND TRANSFER PUMP
401
LO
144
LO
LO
244
LO
LO
344
LO
A
B
C
143
243
343
142
LO
LO
TO UNIT 4
242
LO
LO
TO UNIT 4
342
LO
LO
TO UNIT 4
– TURB. DRIVEN PUMPS
– REQUIRES DC FOR LUBRICATION COOLING
– REQUIRES SERVICE WATER FLOW (CITY WATER)
401
TO COND. TRANSFER PUMP
400
LO
FROM CRS
CST UNIT 4
250,000 GAL.
DC SOLENOID OPERATED AIR VALVES
CV2831
"B"
CV2816
"A"
141
LO
140
139
LO
CV2832
"B"
CV2817
"A"
241
LO
240
239
LO
CV2833
"B"
CV2818
"A"
141
LO
340
339
LO
NC
FC
POSITION INDICATED IN CONT. RM
CONTAINMENT
MAIN FW LINES
TO SGC
TO SGB
TO SGA
FAILS AS IS
DC SOV MODULATING
ATM
T_A
69
NC
3705
83
82B
82A
LO
LO
LO
LO
FROM UNIT 4
ATM
T_B
66
3706
85
84B
84A
LO
LO
FROM UNIT 4
ATM
T_C
68
3707
87
87B
86A
LO
LO
FROM UNIT 4
B TRAIN
MOV1403
NC FA
119
A OR B TRAIN
MOV1404
NC FA
219
A TRAIN
MOV1405
NC
FA
319
AC OPERATED FAIL AS IS.
CONTAINMENT
MAIN STEAM LINES
– STEAM ADMISSION TO AFWS TURBINES (UNIT 3 SHOWN)
LEGEND:
FO FAIL OPENED
FC FAIL CLOSED
FA FAIL AS IS
NORMALLY OPENED
NORMALLY CLOSED
MOTOR OPERATED
AIR OPERATED

Auxiliary Feedwater System

Turkey Point (Units 3 and 4)
Figure 1

20. YANKEE ROWE 2 AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

YANKEE ROWE

AUXILIARY FEEDWATER SYSTEM

X.17.1 System Description

X.17.1.1 Configuration and Overall Design

The auxiliary feedwater system (AFWS) is designed to supply water to the steam generators for reactor coolant system decay heat removal when the normal feedwater system is not available. The AFWS is not normally used for other plant operations such as startup or shutdown. The system can also be used for performing hydrostatic tests during plant shutdown. A dedicated operator for initiating flow for this system is available with direct communication with the control room operator. The auxiliary feedwater (emergency boiler feed pump-EBFP) must be started locally and four normally closed manual valves in parallel in the EBF pump discharge lines must locally opened. After starting the pump, the flow can be controlled from the control room. However, the dedicated operator remains on station even if flow is being controlled from the control room.

The AFWS is shown in simplified form on Figure 1 attached. The system consists of a steam turbine driven positive displacment pump, with steam being supplied from each steam generator into a common header to the pump turbine. Discharge from the pump feeds into a common header which supplies each of four steam generators via the main feedwater piping. Each of the AFWS lines containsa normally closed manual isolation valve. The pump capacity is ≥80 gpm at 1200 psi and takes suction from a 30,000 gallon Demineralized Water Tank (DWT).

A secondary source of water is available from a 135,000 gallon Primary Water Storage Tank (PWST). Water from the 135,000 gallon tank is gravity fed to the 30,000 gallon demineralized water tank by opening one manual valve or directly fed to pump suction by opening a different manual valve. Level indication from the 30,000 gallon and 135,000 gallon tanks are provided in the control room, with high and low level alarms in the control room for the 30,000 gallon tank.

A backup method of supplying feedwater to the steam generators in the event of failure in the AFWS is the plant's primary coolant system charging pumps with total capacity of ~100 gpm (33 gpm/pump). Two of the pumps have variable speed motors. The system is connected permanently by a spool piece that connects to the main feedwater header. The operation of ten manual valves (two drains and eight isolaton) is required to initiate flow from this source. The water supply to the charging pumps is the 135,000 gallon Primary Water Storage Tank.

The High Pressure Safety Injection and Low Pressure Safety Injection pumps provide another backup method of supplying feedwater to the steam generators. Flow from this source is obtained by the operation of the same manual valves used when the charging system is the source, plus the operation of one of two redundant motor operated valves (MOV). Flow is then directed to the steam generators through the same permanently connected spool piece used for the charging pump path as described above. The flow available from this source is 200 gpm per train (three trains available).

AFW flow is controlled by the normal feedwater control valves in the main feedwater (MFW) lines to the steam generators. The preferred system to be used upon demand is the steam driven turbine pump (AFW) system. The charging pumps or the S.I. pumps are backups to the AFW system. The minimum AFW flow required for decay heat removal is approximately 80 gpm.

The turbine driven pump steam admission valve is a manual valve, which is in the auxiliary steam header. The auxiliary steam header is isolated on receipt of a containment isolation signal by operation of an air operated trip valve. Capability is provided to override the containment isolation signal from the control room. The trip valve also closes on loss of air pressure. An alternate supply of nitrogen is provided (in tanks) in the event of loss of the normal air supply. A number of normally open isolation valves are also located in the header between the admission valve and the trip valve that feeds steam to various steam auxiliary systems.

X.17.1.2 Component Design Classification

The steam piping and primary piping (charging, SI systems) are non-seismic systems, Safety Class 2, classified in accordance with ANSI 18.2, which requires either safety Class 2 or 3 piping. Control and Instrumentation Systems are non-Class IE.

X.17.1.3 Power Sources

Power for the charging pumps and MOVs is supplied from separate nonsafety 480V AC buses, which are capable of being fed by the emergency 480V AC buses by remote manual operation of circuit breakers. The injection pumps are connected to the 480V emergency buses. Offsite power normally feeds the emergency buses. Diesel generators are automatically connected to the emergency 480V AC buses on loss of offsite power.

The plant electrical bus arrangement consists of three independent divisions of 2400V AC buses, one bus fed by one offsite line, a second fed from an independent offsite line, and the third fed from the unit generator. Capability exists to manually transfer from one supply to the other. The three 2400V AC divisions then feed three independent 480V AC through transformers.

The instrumentation and control power is 120V AC from an inverter connected to the 125V DC battery supply.

X.17.1.4 Instrumentation and Controls

X.17.1.4.1 Controls

The water level for each steam generator is controlled manually from the control room by the feedwater controllers that normally are used in the main feed system lines. Steam generator water level can also be controlled locally at the controllers. All MOVs can be remote manually operated from the control room. The charging pumps and SI pumps can be started from the control room.

X.17.1.4.2 Information Available to the Operator

The following indications are available in the control room:

1. Level Indication - 30,000 gallon demineralized water tank
2. Level Indication - 135,000 gallon PWST
3. Flow to steam generators when SI system used
4. Charging pump discharge pressure
5. Steam generator water level
6. Steam generator steam pressure

X.17.1.4.3 Initiating Signals for Automatic Operation

The AFWS initiation is manual. (See section 17.1.1 for manual operation)

X.17.1.5 Testing

Steam Turbine System

The steam turbine is tested every 15 days and operated for 15 minutes. The discharge pressure is monitored to verify rated output (950 psi).

In addition to the operational test, the valve lineup of the system is verified.

The SI system is tested weekly on a staggered basis. The flow is recirculated to the supply tank and pump current is monitored (vibration tests are performed monthly for both the AFWS and SI system). In addition, at the completion of the operational test, valve position of the system is verified.

X.17.1.6 Technical Specifications

The AFWS must be operable or the unit must be in hot standby in one hour and hot shutdown in next 12 hours.

X.17.1.7 Additional Information

The AFW system is manually actuated, however, approximately one hour of steam generator water inventory is available subsequent to loss of feedwater and reactor shutdown.

The offsite power is exceptionally reliable, having experienced only one outage in 19 years of operation.

No challenges to the AFW system have been made during the entire operational history.

X.17.2 Reliability Evaluation Results

X.17.2.1 Dominant Failure Modes

Failure modes of the AFWS were assessed for three types of initiating transients. The dominant failure modes for each transient type are discussed below.

Loss of MFW with Offsite Power Available

The dominant failure mode of the AFWS for this transient results from a set of human errors. The first human error, which causes the unavailability of the AFWS, is the inadvertent closure of one of six manual valves in the steam supply line to the AFWS pump turbine. Upon a demand for the AFWS, the operator has up to an hour to detect this fault and correct it (i.e., open the valve). An alternative for the operator is to manually open the valves from the charging pumps and supply water to the steam generators from these pumps. Thus, the dominant failure mode is the combination of a human error inadvertently closing one of the steam supply line valves and the error of failing to reopen the valve, or realigning the charging pumps, within about one hour after a demand on the AFWS.

Loss of MFW with Only Onsite AC Power Available

AC power dependencies were considered as potential faults for this analysis. It was concluded that the dependence on onsite power instead of offsite power does not significantly alter the results of the assessment. Thus the dominant failure mode of combinations of human errors before and after the transient event is considered to be dominant for this transient also.

Loss of MFW with Only DC Power Available

For this event, the probability of AFWS failure is reduced to the probability of failure of the steam driven pump train. The dominant failure mode within this train is failure to provide steam to the turbine, caused by the inadvertent closure of any 1 of 6 valves in the steam supply line, coupled with failure to reopen the closed valve(s) within approximately one hour after a demand on the AFWS.

X.17.2.2 Principal Dependencies

Within this plant, the principal dependency is the requirement for human actions, such as valve manipulations, to start the AFWS or the backup systems such as the charging pumps or the safety injection pumps. No other dependencies of significance were identified in this evaluation.

X.17.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.17.3.1 Short-Term

1. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation of flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiaton and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5. Recommendation - The AFW surveillance tests should require that the normally closed manually operated valves in the connection between the charging pumps/safety injection pumps and the AFW system be cycled each quarter.

X.17.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system design at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator actions, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shutdown and cooled down then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Postion 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic test on one AFW system train, and there is only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communiciation with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.17.3.3 Long-Term

Long-term recommendations for improving System are as follows:

1. Recommendation - At least one AFW system pump, its associated flow path and essential instrumentation should be capable of being initiated from the control room and being operated independently of any alternating current for at least two hours. Conversion of direct current to alternating current is acceptable.

2. Recommendation - Initiation of AFW flow (including flow from the backup systems-charging/SI) to the steam generators requires several local manual operator actions outside the control room. Even though there is a reasonable time period (up to one hour before the S/G's will boil dry) for operator action and a dedicated operator, the licensee should improve the reliability of initiating AFW flow by providing the capability to start the pumps and open the valves of the AFWS by operator action from the control room. Local manual operation capability should be retained as a backup to remote manual operation capability.

3. Recommendation - A pipe break in the Main Feedwater header upstream of the control valves could cause loss of all AFW flow to all steam generators since the AFW pump and the charging/SI pumps connect to this header. The licensee should evaluate the consequences of a pipe break in this section of the MFW header and 1) determine any system design changes or emergency procedures necessary to detect and isolate the break and direct the required AFW flow to the steam generators before they boil dry or 2) describe how the plant can be brought to a safe shutdown condition by use of other available systems following such a postulated event.

4. Recommendation - The air operated trip valve in the auxiliary steam header which supplies steam to the turbine driven AFW pump closes upon receipt of a containment isolation signal. The

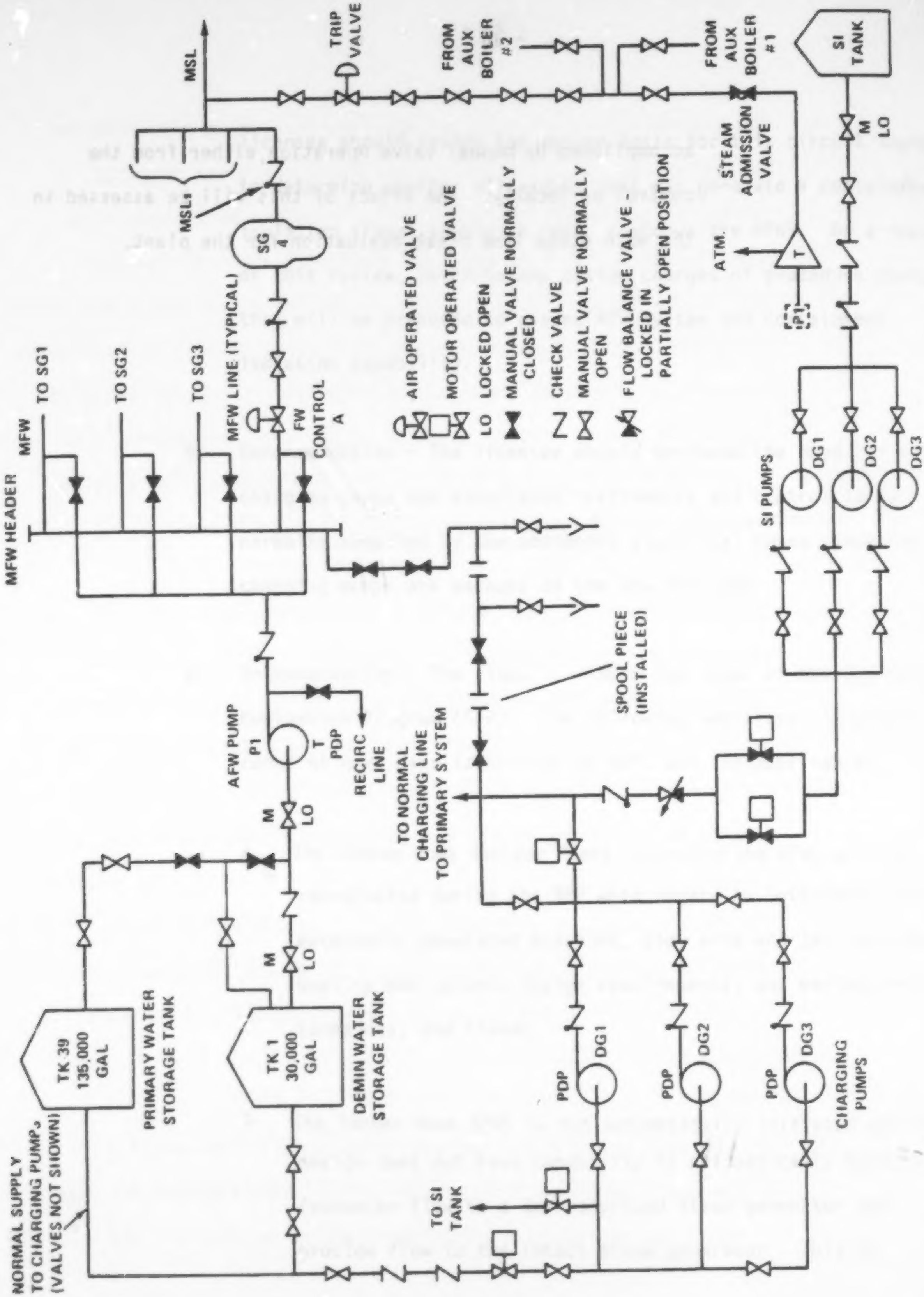
licensee should review the design basis for this circuit logic to determine whether all events that can generate a containment isolation signal should in fact, shutdown the AFWS. As a result of this review, describe any design changes of procedure changes that will be proposed to assure AFW system and containment isolation capability.

5. Recommendation - The licensee should evaluate the need for the charging pumps and associated instruments and control to be normally supplied by the emergency electrical buses since the charging pumps are backups to the one AFW pump.

6. Recommendation - The plant is within the scope of the Systematic Evaluation Program (SEP). The following additional long term concerns have been identified by SEP, and are applicable.

   a. The Yankee Rowe Nuclear Plant including the AFWS will be reevaluated during the SEP with regard to internally and externally generated missiles, pipe whip and jet impingement, quality and seismic design requirements, and earthquakes, tornadoes, and floods.

   b. The Yankee Rowe AFWS is not automatically initiated and the design does not have capability to automatically terminate feedwater flow to a depressurized steam generator and provide flow to the intact steam generator. This is accomplished by manual valve operation,either from the control or locally. The effect of this will be assessed in the main steam line break evaluation for the plant.


MSL
TRIP VALVE
MSLs
SG 4
FROM AUX BOILER #2
FROM AUX BOILER #1
STEAM ADMISSION VALVE
SI TANK
M LO
ATM.
T
P1
MFW HEADER
MFW
TO SG1
TO SG2
TO SG3
MFW LINE (TYPICAL)
FW CONTROL A
AIR OPERATED VALVE
MOTOR OPERATED VALVE
LO LOCKED OPEN
MANUAL VALVE NORMALLY CLOSED
CHECK VALVE
MANUAL VALVE NORMALLY OPEN
FLOW BALANCE VALVE LOCKED IN PARTIALLY OPEN POSITION
SI PUMPS
DG1
DG2
DG3
AFW PUMP
P1
T
PDP
RECIRC LINE
TO NORMAL CHARGING LINE TO PRIMARY SYSTEM
SPOOL PIECE (INSTALLED)
M LO
M LO
NORMAL SUPPLY TO CHARGING PUMPS (VALVES NOT SHOWN)
TK 39 135,000 GAL
PRIMARY WATER STORAGE TANK
TK 1 30,000 GAL
DEMIN WATER STORAGE TANK
PDP
DG1
PDP
DG2
PDP
DG3
CHARGING PUMPS
TO SI TANK

Auxiliary Feedwater System
Yankee Rowe

21. ZION 1 AND 2 AUXILIARY FEEDWATER SYSTEM

ENCLOSURE 1

ZION UNITS 1 AND 2
AUXILIARY FEEDWATER SYSTEM

X.18.1 System Description

X.18.1.1 Configuration, Overall Design

A simplified drawing of the Zion Auxiliary Feedwater (AFW) System is shown in Figure X.18-1. The design configuration is the same for both units. The system consists of two motor-driven pumps,each with a capacity of 450 gpm at 3099 feet head and one steam turbine driven pump with a capacity of 900 gpm at 3099 feet head. The motor-driven pumps feed a common header,which in turn feeds all four steam generators through motor-operated throttle valves that limit the flow to 105 gpm per steam generator. The turbine-driven pump feeds a common header,which feeds all steam generators through motor-operated throttle valves,which limit the flow to 105 gpm per steam generator. The normal auxiliary feedwater supply is the condensate storage tank,which supplies a common header that feeds all three pumps. There are cross-connections between all pump discharge trains,as shown in Figure X.18-1 with the cross-connection between the motor-driven pump and turbine-driven pump trains normally closed; all other valves are in a normally open-fail open or fail-as is position, as shown in the figure.

The AFW system was evaluated for high energy line breaks, with and without a single active failure, in the main steam system, main feedwater system and AFW system. In all cases considered, auxiliary feedwater can be put into the steam generator within 20 minutes of the break or indication of a pipe break, through one of the unaffected trains. There was only one place in the pipe chase where a break in a auxiliary feedwater line might damage the other auxiliary feedwater line. However, since the lines are the same size, this would not cause a break in the other line. Consequently, in case of a break in an auxiliary feedwater line, water can still be supplied to the steam generator(s). (Section 18.2.2 discusses an additional pipe break concern).

Source of Water

The auxiliary feedwater system has five sources of water. The primary source of water is the non-seismic condensate storage tank for the unit. This tank has a capacity of 500,000 gallons of which 170,000 gallons are administratively dedicated to the auxiliary feedwater system. These 170,000 gallons will provide a minimum of eight hours of water to the steam generator,with the steam being dumped to atmosphere. Valves in the supply line to the AFW system are normally open. The tank is not tornado missile protected, but the lines are buried, so some protection from tornado missiles is provided.

The secondary source of water is the condensate storage tank for the other unit. The cross connection has a normally closed manually operated valve, but there are other means of transferring water from tank to tank, if needed.

The backup source (and one of the long-term sources of water) is the seismic Category I service water system, which can supply both units. The source of service water is Lake Michigan. A common header from the service water system serves all pumps. In order to put the system into operation as a backup for the AFW system, motor-operated valves have to be opened. There is a manually-operated cross tie with the other unit's service water system. There is the possibility for minor flooding in the auxiliary building when the service water system is used for long-term cooling. There are some vents in the service water system which will discharge about 20 gpm, if they are not closed when the service water system is used to supply the AFW system.

The other long-term source of water is the makeup demineralizer system, which is not designed to seismic Category I requirements. Although this system feeds directly to the condensate storage tanks at a maximum rate of 1200 gpm, it would take between 10-30 minutes to put the system in operation manually. The water source for this system is also Lake Michigan.

X.18.1.2 Components - Design, Classification

Table X.18-1

	Environmentally Qualified (line break)	Design Classification	Seismic Category
Steam Driven Pump and Turbine	Yes	1	I
Motor Driven Pumps and Motors	Yes	1	I
Piping - In Auxiliary Building	Yes	1	I
from Condensate Storage Tank to Auxiliary Building		3	N.S.*
Valves -	Yes	Same as piping	
Power Supplies	Yes	1	I
Instrumentation	No	3	N.S.*
Controls	Yes	1	I
Condensate Storage Tank	-	3	N.S.*
Service Water System	-	1	I
	Environmentally Qualified (line break)	Design Classification	Seismic Category
Make-up Demineralization System	-	3	N.S.*
Main Steam Supply for Turbine Driven Pump including Valves	-	1	I

*N.S. (non-seismic) - not designed to meet seismic Category I requirements.

X.18.1.3 Power Sources

The motor-driven pumps are powered from separate emergency diesel-generators; i.e., one pump to each diesel-generator. The eight motor-operated throttle valves take power from the diesels. Two out of four throttle valves for the motor-driven pump trains are run off one diesel-generator. The other two are on another diesel-generator. The four on the turbine-driven pump train are divided the same way. All motor-operated valves are in the open position and fail as-is.

The steam turbine-driven pump takes its steam from steam generators A and D through a common header, a normally open, motor operated, fail-as-is valve, and an air-operated, fail-open, control valve. The solenoid for the air-operated valve is powered from the direct current buses.

The alternating current power for the instrumentation is derived from the direct current buses, which take power from the onsite alternating current system, or from the station batteries through an inverter.

The power for the service water pumps comes from the alternating current power system. Since the motor-driven and turbine-driven pumps have lube oil coolers, the cooling from these pumps comes from the service water system. If the plant were to experience a station blackout (loss of offsite and onsite alternating current power), the licensee estimates that the turbine-driven pump would last a minimum of 15 minutes without cooling and the steam generators would boil dry approximately 30 minutes after the auxiliary feedwater pump stopped. We believe that this condition is unacceptable, and adequate lube oil cooling for the turbine-driven pump should be provided which is independent of alternating current. (See Recommendations).

X.18.1.4 Instrumentation and Controls

X.18.1.4.1 Controls

The AFW system is an entirely automatic system, except for the switching of the sources of water. This switching must be done either in the control room or locally, depending on the source. The manual controls located in the control room are the pump on-off switch, valves open-close switches, and the throttle valve control switch, which is used for steam generator level control. The controls at the remote shutdown panel and local AFW operating station are the same as those in the control room, except for the lube oil pump controls. which are only located at the remote shutdown panel which is also the local AFW operating station.

X.18.1.4.2 Information Available to Operator

The indication available to the operator is as follows:

1. Alarms (Control Room Only)
    a. Low suction pressure (common all pumps);
    b. Auxiliary feed pump not available (common all pumps);
    c. Condensate tank low and high level;
    d. Low lube oil pressure trip (common all pumps);
2. Indicators (Control Room and Remote Shutdown Panel, except as noted)
    a. AFW flow to each generator;
    b. Condensate storage tank level; (Control Room and Rad Waste Panel)
    c. AFW pump running;
    d. Valve indications (open-close);
    e. AFW pump turbine inlet steam pressure; (Control Room only)
    f. AFW pump discharge pressure;
    g. AFW pump motor current.

X.18.1.4.3 Initiating Signals for Automatic Operation

The AFW system flow is automatically initiated on any of the following signals:

1. Steam Generator low-low level (10% narrow range)
    a. Motor-driven pumps start on one steam generator low-low level;
    b. Turbine-driven pump starts on two steam generators low-low level.
2. Loss of offsite power;
3. Safety injection signal;

The AFW system can also be initiated by the operator from the control room and/or the Remote Shutdown Panel.

X.18.1.5 Testing

The AFW system is tested on a monthly basis, one train at a time. The procedure tests all components of the system. It is composed of closing the throttle valves for the train being tested, and running the system in the recirculating mode. The throttle valves are then opened to allow a flow of 105 gpm to the steam generators. All valves, including the service water system valves and the throttle valves, are stroked monthly. At the same time the monthly tests are being performed, a vibration test is performed on the pumps.

When a train is being brought back into service after a maintenance outage, the above tests are performed, except that the vibration test is deleted.

X.18.1.6 Technical Specifications

Table X.18-1 details the limiting conditions for operation and surveillance requirements for the Zion Station AFW system. Problem areas identified during our review are that (1) one pump can be out of service indefinitely with no action taken, and (2) the licensee does not consider the instrumentation as part of the system. Thus the instrumentation for one or more pumps could be out of service while the plant is allowed to operate. (See Recommendation).

X.18.2 Reliability Evaluation

X.18.2.1 Dominant Failure Modes

The Zion AFW system was analyzed for the following cases:

a. Loss of main feedwater with offsite power available;

b. Loss of main feedwater with onsite alternating current power available;

c. Loss of main feedwater with only direct current power available.

The results of the analysis are summarized below.

ZION TECHNICAL SPECIFICATIONS

LIMITING CONDITION FOR OPERATION

Auxiliary feedwater pump system

A. Two of the three auxiliary feedwater pump systems shall be operable whenever the reactor is going from cold shutdown to hot standby.

B. Two of the three auxiliary feedwater pump systems shall be operable whenever the reactor is in hot standby or operating except as specified in 3.7.2.C.

C. From and after the date that two of the three auxiliary feedwater pump systems are made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days provided that during these 7 days the remaining auxiliary feedwater pump system shall be operable.

D. If these conditions cannot be met the reactor shall be brought to the hot shutdown condition within four hours. After a maximum of 48 hours in the hot shutdown condition, if the system is not operable the reactor shall be brought to the cold shutdown condition within 24 hours.

SURVEILLANCE REQUIREMENT

Auxiliary feedwater pump system
(Table 4.7-2)

A. Surveillance and testing of the auxiliary feedwater pump systems shall be performed as follows:

1. The auxiliary feedwater pumps shall be started manually from the control room each month. Performance will be acceptable if the pump starts upon actuation, operates for at least 10 minutes on recirculation flow, and the discharge pressure and flow are within ±10% of a point on the pump head curve.

2. The service water power operated supply valves to the auxiliary feedwater pumps shall be stroked manually from the control room each month. Performance will be acceptable if valve motion is indicated upon actuation.

B. Not Applicable.

C. When it is determined that two of three auxiliary feedwater pump systems are inoperable, the one remaining system shall be started immediately and daily thereafter including the associated standby AC and DC power supplies (See Section 4.15.1.B.2 and 4.15.1.E.1).

D. Not Applicable.

Loss of Main Feedwater with Offsite Power Available

The unavailability of the AFW system for this case is dominated by the following failures:

a. maintenance being performed on the second manual valve in the supply line from the condensate storage tank + failure to manually initiate service water to supply the system upon demand, and,

b. turbine-driven pump train out for maintenance over an extended period of time + testing of one of the motor-driven pumps.

The first of these failures disables the supply water to the AFW system. The second disables AFW system discharge to the steam generators in the following manner. Maintenance on the turbine-driven pump train results in four lines to the steam generator not receiving flow; testing for the motor-driven pump train requires closing of the motor-operated valves in the other four lines to the steam generators, thus closing off all discharge flow to the steam generators for the period of the test.

Loss of Main Feedwater with Onsite Alternating Current Power Available

The AFW system was analyzed assuming loss of offsite power, considering possible loss of one of the diesel generators. Failure of a diesel-generator has no significant effect on system reliability. The dominant failure modes appear to be similar to those discussed in the previous case.

Loss of Main Feedwater with Only Direct Current Available

For this case, neither offsite nor onsite alternating current power are available. The system may not successfully operate for an extended time period without alternating current power due to the fact that the turbine-driven pump lube oil is dependent upon the alternating current powered service water system for cooling. Without alternating current power, lube oil cooling is lost, which could result in failure of the only operable AFW pump.

In the short-term (≤ 45 minutes), the turbine-driven pump train unavailability is dominated by maintenance, test, and single hardware failures.

X.18.2.2 Dependencies

In addition to the dependence of the turbine-driven pump train on alternating current power discussed above, one locational dependence was identified. The two motor-driven pumps and their associated motor-operated valves on the suction side are in a common location. There is only a short barrier (about six feet tall) between this cell and the cell containing the turbine-driven pump and its suction motor-operated valve. Thus, there are potential location-dependent interactions in this system. A high energy line (steam line to the turbine-driven pump) passes through this space. Although the pumps are qualified for a steam environment, it does not appear that the motor-operated valves are (See Recommendations).

X.18.3 Recommendations for this Plant

The short-term recommendations identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.18.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumention can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

3. Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

   The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

4. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

5. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done,

the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

6. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:
   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

7. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to intiate the AFW system from the control room.

8. Recommendation GS-8 - The licensee should propose modifications to the Technical Specifications to include a Limiting Condition of Operation when the condensate storage tank level falls below the 170,000 gallon level, considering both one- and two-unit operation.

X.18.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

> "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.
>
> The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.18.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

   Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

   The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

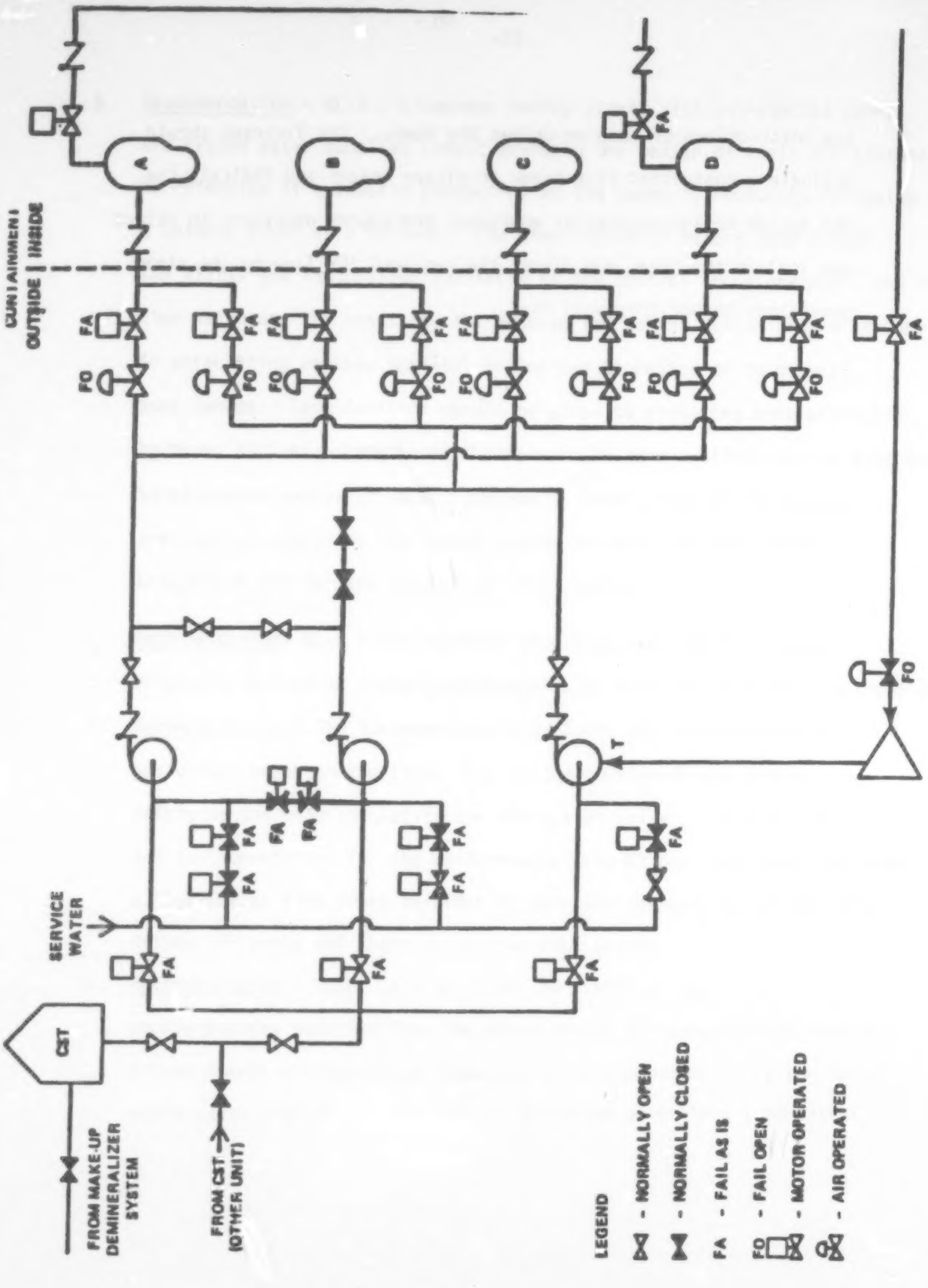
3. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

4. Recommendation- GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

5. Recommendation - The licensee should evaluate the consequences of a postulated break in the steam line to the turbine-driven AFW pump to determine the need to qualify the AFW system valves, valve actuators, and instrumentation for the environmental conditions resulting from such a high energy line break in order to maintain operability of the motor-driven AFW pumps and their associated flow trains.

6. Recommendation - There are no valves in either of the common headers supplied from the motor-driven AFW pumps or the turbine-driven pump to all four steam generators. A pipe break in either header could cause loss of all AFW flow to all steam generators from either

the motor-driven or turbine-driven AFW pumps. The licensee should evaluate a postulated pipe break in either header and indicate the AFW system design changes or emergency procedures necessary to detect and isolate the break and direct the required AFW flow to the steam generators before they boil dry.


CONTAINMENT
OUTSIDE | INSIDE
A
B
C
D
FA
FO
T
SERVICE WATER
CST
FROM MAKE-UP DEMINERALIZER SYSTEM
FROM CST (OTHER UNIT)
LEGEND
- NORMALLY OPEN
- NORMALLY CLOSED
FA - FAIL AS IS
FO - FAIL OPEN
- MOTOR OPERATED
- AIR OPERATED

Auxiliary Feedwater System

ANNEX 1 TO APPENDIX X

May 4, 1979

Attachment

As part of its ongoing review of the Three Mile Island Unit 2 accident, the staff finds that it need additional information regarding the auxiliary feedwater systems (AFWS). This information, as outlined below, is required to evaluate AFWS reliability for Combustion Engineering (CE) and Westinghouse (W) designed pressurized water reactors. The requested information is in addition to that requested in the IE Bulletins, and should be brought to the meeting scheduled with the staff on May 8 thru May 12, 1979.

Written system description (as built), including:

- List of support systems for auxiliary feed system operation (both electric and steam)
- Water supplies for AFWS (primary and backup)

Current operating procedures and test and maintenance requirements, including:

- All LCO's for AFWS, main FW system and related support systems
- Listing of operator actions (local and/or control room) and timing requirements for such actions
- Procedure for reinitiating main feedwater flow

As built P&ID's with symbol keys including condensate and steam side

Legible equipment layout drawings, including:

- Isometrics, if available
- Identification of inhibits preventing accessibility to AFWS components and related electrical equipment

Relevant control systems description, including:

- Schematic or logic control diagrams
- Listing of actuation signals/logic and control
- MSIS logic for isolating AFWS, if installed
- Electric power dependencies
- All "readouts" available in control room for AFWS operation

AC and DC Power

- One line diagrams (normal and emergency power supplies)
- Divisional designation, e.g., Train A, Train B, requirements on all AFWS components and support systems
- List of normal valve states and loss-of-actuation power failure position

Operating Experience, including

- Number of main feedwater interruptions per year experienced to date for each unit
- Number of demands on AFWS per year to date (test and actual) for each unit
- Summary of AFWS malfunctions, problems, and failures

Provide available reliability analyses

Steam generator dry-out times (assuming loss of all feedwater flow, with 100% initial power, with reactor trip, no line breaks)

System design bases, including:

- Seismic and environmental qualification
- Code and quality, QA

best copy available

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The purpose of this report is to summarize the results of the Bulletins & Orders Task Force generic review of feedwater transients, small break LOCAs, and other Three Mile Island, Unit 2 types of events in Westinghouse-designed operating plants and to confirm the bases for their continued operation. The results of this evaluation are presented in this report in the form of a set of findings and recommendations in each of the principal review areas.

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